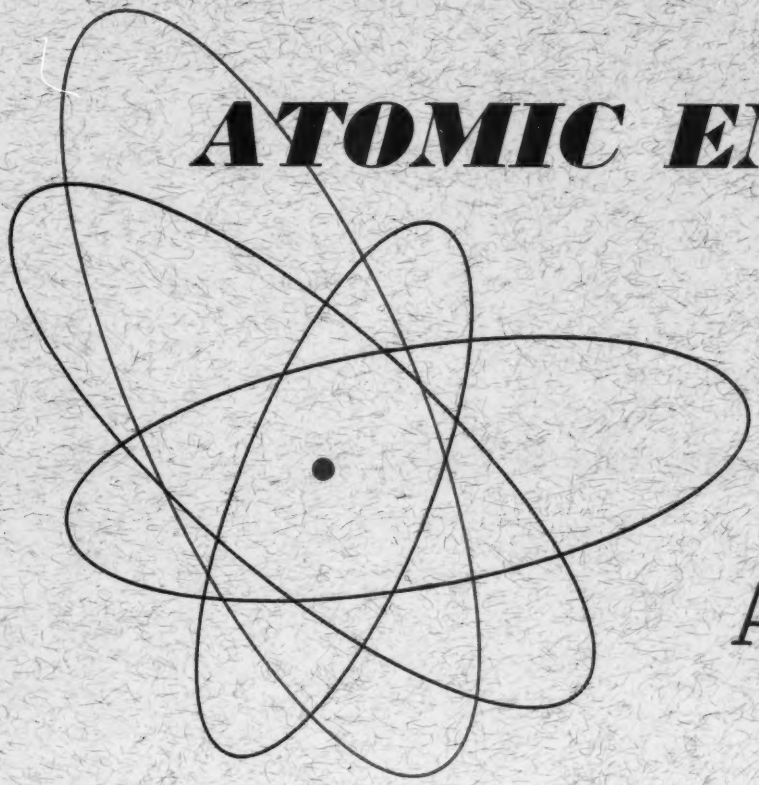
Volume 6, No. 1
August, 1960

THE SOVIET JOURNAL OF

ATOMIC ENERGY



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THE SOVIET JOURNAL OF

ATOMIC ENERGY

A translation of ATOMNAYA ÉNERGIYA,
a publication of the Academy of Sciences of the USSR

(Russian original dated January, 1959)

Vol. 6, No. 1 August, 1960

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ANNOUNCEMENT

In order to avoid senseless duplication, and speed the availability in English translation of original Soviet research, we shall, starting with the present issue, no longer translate those items in the "Scientific and Technical News" section which are reports concerning Western research and development and which are essentially abstracts of material originally published in English-language periodicals.

Furthermore, we shall not reprint material such as, the address by G. T. Seaborg which appeared on page 21 of this issue of Atomnaya Energiya, and which is readily available in English from other sources.

We shall continue to present the complete translated text, including all tabular and illustration material, of all research and review articles, editorial notes and announcements, book reviews, bibliographies, etc., and all news items of other than United States or British Commonwealth origin.

The Table of Contents will continue to list all material which appeared in Atomnaya Energiya, but those items which originated in the English language and are therefore not included in the translation will be enclosed in brackets. Whenever possible, the source from which English copies of the omitted reports can be obtained will be given.

Consultants Bureau Enterprises, Inc.

COMPLEX UTILIZATION OF URANIUM ORES

B. V. Nevskii

In recent years uranium deposits with rich ores have become almost completely exhausted. At the same time, the demand for uranium is increasing, and therefore poor uranium ores are being exploited. In regard to the profitable treatment of these ores, considerable importance attaches to their complex exploitation with the extraction of other constituents. The complex exploitation of gold-uranium and uranium-vanadium ores is being carried out on an extensive scale, and the problems involved in the recovery of uranium and accompanying constituents from uranium-containing phosphorites, coal and shale, uranium-thorium ores, as well as uranium-copper, uranium-pyritic, zirconium-uranium, niobium-tantalum-uranium and uranium-molybdenum ores are being resolved in a positive manner. The increased interest in the complex exploitation of uranium ores enables some new forms of mineral raw materials to be included in the industrial scope.

The increased demand for uranium and the exhaustion of rich deposits are giving rise to the necessity of exploiting poor uranium ores, in the treatment of which the economic factors of the process employed are decisive, since they determine the cost of the uranium produced and, consequently, the expediency of working such ores.

One method of reducing the cost of the uranium produced from poor ores is their complex exploitation; that is to say, the organization of the recovery of certain constituents contained in the uranium ores.

The occurrence of two valencies and the tendency to the formation of complexes result in a diversity of the mineral forms of uranium and in the presence of uranium in ores of widely differing types, in consequence of which uranium, from the geochemical point of view, is a scattered element. Many uranium ores contain other valuable constituents, in addition to uranium: gold, phosphorus, sulfur, molybdenum, copper, cobalt, bismuth, niobium, tantalum, fuel, and others. In a number of cases, the content of these elements is not very high, but in others it may be quite considerable, and then these constituents become of independent value, while uranium may be the accompanying constituent.

The recovery of the accompanying constituents may increase the economic factors in the treatment of uranium ores, extend the raw material sources of uranium, and provide industry with a considerable quantity of important metals and other constituents.

Due to the large-scale production of such useful minerals as gold-bearing ores, phosphorites, coal, etc., uranium may be recovered from them profitably and in considerable amount, even if its content is quite low.

The reserves of uranium in complex ores according to the figures for 1955-1957 are quite considerable (Table 1). The extraction of the accompanying constituents may be carried out in accordance with the following principal schemes:

1) Separation in the concentrate by methods of mechanical beneficiation, before or after extraction of the uranium;

2) Separate extraction of the accompanying constituents by means of special reagents, before or after extraction of the uranium;

3) Joint extraction along with the uranium, and individual separation from the solutions, before or after removal of the uranium;

4) Separation of the accompanying constituents by the use of thermal or other special processes, before or after the extraction of the uranium.

These operations ought not substantially to interfere with, or increase the difficulty of, the extraction of the uranium, or increase its loss. The extraction of uranium from ores in which it is an accompanying constituent ought not to interfere with the technology of the extraction of the basic constituents.

The following processes have been developed and perfected, and are being used in industry for the complex treatment of uranium-bearing ores.

Gold—Uranium Ores [1-3]

These ores are the most important of the complex uranium ores (Table 1). At the present time, industrial deposits are known only in South Africa (Witwatersrand). They have been worked already for more than

Table 1. Uranium Reserves in Complex Ores.

Country	Kind of ore	Ore reserves, millions of tons	Content in ore		Uranium reserves, thousands of tons
			Uranium, %	Accompanying constituents	
South Africa	Gold-bearing conglomerates	1000	0.0285	5–10 g/ton gold	285
United States of America	Phosphorites	1000	0.1	20–25% P_2O_5	100
	Carnotite sands	63.3	0.212	up to 1–2% V_2O_5	135
Canada	Brannerite-bearing conglomerates (uranium - basic)	205	0.09	up to 0.05% ThO_2	185
Sweden, USA, and others including Sweden*	Carbonaceous shales	18,000	0.001–0.25	15–30% combustible matter	1800
		500	0.02	25% combustible matter	100

*As in original—Publisher.

60 years and provide more than 50% of the output of gold in foreign countries. The occurrence of uranium in these ores was first discovered in the twenties, but the practical significance of this uranium was not realized until after 1945. In South Africa, USA, United Kingdom, and Canada from 1946 to 1951, intense scientific research and experimental work was carried out on the extraction of uranium from the tailings of the gold mines of Witwatersrand.

In 1949-1951, two experimental plants were constructed, and in 1952 the first industrial plant for the extraction of uranium from the tailings, after the extraction of the gold, began to operate; at the present time, 17 such plants are in operation in South Africa, with a total throughput of about 50,000 tons of ore daily, which is about 1/3 of all the ores worked for the extraction of gold. Since 1957 these plants have produced more than 3000 tons of uranium in chemical concentrates (commercial uranous-uranic oxide). Figure 1 shows a typical flow diagram of a plant for the extraction of uranium from gold-bearing ores.

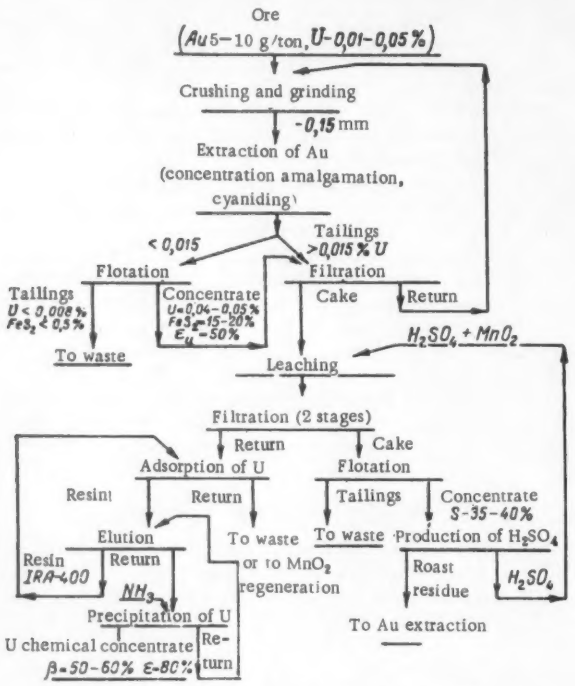

Fig. 1. Flow diagram of complex treatment of gold-uranium ores in South Africa.

The uranium is extracted both from the tailings of working concerns and from refuse dumps; the latter are worked by hydraulic excavators and the material sent to the uranium plant by hydraulic conveyors. The tailings containing more than 0.015–0.020% uranium are treated directly for leaching out the uranium (after preliminary filtration to remove cyanide residues and other harmful impurities, as well as excess water). The poorer tailings (0.008–0.015% uranium) are first sent to the flotation plant, where about 50% of the uranium and about 90% of the pyrite are removed, after which the concentrate passes to the uranium leaching plant. Leaching is by sulfuric acid, with manganese dioxide added as oxidizing agent, in the cold or in slight warmth (up to 50°C) for 16–24 hours at pH = 1–1.5. The uranium-containing solutions are separated by double filtration on revolving filters, whereupon the uranium is extracted from them by adsorption on ion-exchange resins. It should be mentioned that adsorption of uranium on an industrial scale was employed for the first time at the uranium plants of South Africa. After elution, the solution contains about 10 times more uranium and 200 times less impurities than the original solution. By means of ammonia a chemical concentrate is precipitated from it, which, after calcination, contains more than 60% uranium. The chemical concentrates are transported in the form of a thick pulp in tanks from all the plants to a central plant, where they are calcined for the production of oxide, assayed, packed and despatched for further treatment (mainly to USA and partly to the United Kingdom). After extraction of the uranium, the tailings are sent to the flotation plant for the extraction of the pyrite. The pyritic concentrate is utilized on the spot for the production of sulfuric acid used for the leaching of uranium. About 85% of the sulfuric acid is obtained from the pyritic concentrate, the remainder is obtained from elementary sulfur imported from USA.

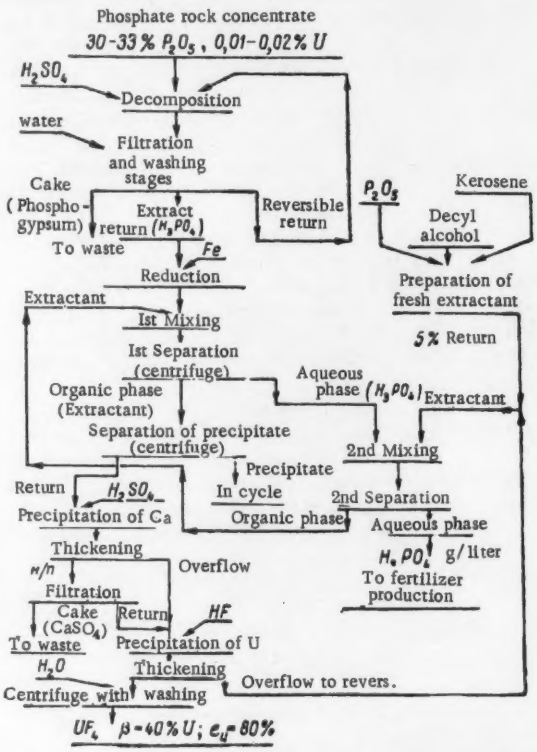

Fig. 2. Flow diagram of complex treatment of the uranium-containing phosphate rock in Florida, USA (plant in Bonnie).

The absence of outlay for mining, crushing, and grinding the ore, the simplicity of the technological scheme, low reagent consumption (altogether about 30 kg/ton ore), the collateral recovery of pyrite for the production of sulfuric acid on the spot, and the cheap local labor ensure a low cost of production of uranium at

South African plants, which is on a level with the cost of uranium from ores with a uranium content of about 0.1–0.2%.

Uranium-Containing Phosphate Rock [4-6]

Uranium in an amount of about 0.01% or more is encountered in many phosphate rock deposits (in Florida, Morocco, and elsewhere). In the organization of the collateral recovery from these sources, it is also possible to obtain a considerable quantity of uranium. At present, the collateral extraction of uranium is being carried out as an experiment at 2 or 3 phosphate rock treatment plants in the USA. About 1000–1500 tons of uranium annually may be produced at these plants in the event of the organization of extraction on an industrial scale.

A flow-diagram of the process employed for the extraction of uranium from phosphate rock is shown in Fig. 2.

The starting operations of the scheme are the usual processes of wet treatment of the phosphate rock with the production of so-called extraction phosphoric acid, containing about 80–90% of the uranium. The uranium is extracted from this acid by means of alkyl phosphates (usually after preliminary reduction by iron). The most effective extractive reagents for uranium in this case are phosphoric acid ethers of the higher alcohols (octylpyrophosphate, dodecylphosphoric acid, etc.) with 4 to 17 carbon atoms in the chain (extraction is better, the greater this number).

Uranium in the tetravalent form is extracted better than in the hexavalent form, the following reaction taking place:

$$2(C_8H_{17}P_2O_7)H_2 + U^{4+} \rightarrow (C_8H_{17}P_2O_7)_2U + 4H^+.$$

Iron, on the contrary, is extracted better in the trivalent form, and therefore reduction of the solution considerably increases the extraction coefficient for uranium (up to 8-12 for an energy equal to 200–300 mv) and diminishes it for iron and a number of other impurities; i.e., it increases the degree of concentration and purification of the uranium. Extraction is carried out in 2-4 stages in countercurrent. Uranium can be precipitated from the organic phase by hydrofluoric acid as a tetrafluoride concentrate. According to this scheme, 1 kg of extracted uranium requires 4-12 kg isooctylpyrophosphate, 15 kg kerosine, 3-4 kg iron, and 1.5 kg 100% hydrofluoric acid.

The chief difficulties in the way of extensive application of this process are the instability of the extractants employed (which are decomposed by hydrolysis and other secondary reactions) and their propensity to form emulsions. To diminish the decomposition of the extractants, the time of their contact with the aqueous phase must be reduced, which is achieved by the use of high-speed dispersion centrifuges (with a dispersion factor of up to 7000), as well as by special centrifugal extractors.

In some plants in the USA, centrifuges with a capacity of 2 m^3 and a throughput of 45 m^3 of solution per hour are used for phase dispersion. Emulsion losses are reduced by the addition of special antiemulsifiers, especially phosphates of high-molecular alcohols (for example, tributyl phosphate).

Research work is also being carried out on the adsorption of hexavalent uranium from extraction phosphoric acid by means of strong-base ion-exchange resins, followed by elution by dilute sulfuric or hydrochloric acid in the presence of reducing agents. There are also a number of patents for chemical methods of separating uranium from extraction phosphoric acid, but so far they have not been used industrially.

Uranium–Vanadium Ores [7-9]

The principal deposits of uranium-vanadium ores are on the Colorado Plateau. Until recently, they were the principal sources of production of uranium in the USA. Originally, these ores were worked for the purpose of extracting vanadium from them. For example, in 1945, about 50% of the world production of vanadium was obtained from these ores. Now, the ores are worked principally for the production of uranium, and only ores with a content of more than 1% vanadium are subjected to a complex treatment.

At the present time, in view of the considerable output of vanadium from other sources, its collateral production is carried out only at 5 plants out of 14 treating the ores of the Colorado Plateau.

Figure 3 shows a typical flow diagram of the complex treatment of uranium-vanadium ores.

Roasting with sodium chloride converts the vanadium into a water-soluble form, sodium vanadate, which makes it easy to separate it from the uranium. The roasting temperature should be of the order of 820-840°C; at a higher temperature, the extraction of the uranium is impaired. For a good extraction of uranium in the subsequent leaching process, it is also necessary to have an excess of vanadium in the mix, this being achieved by returning part of the impure vanadium pentoxide, obtained in the acid leaching of the uranium.

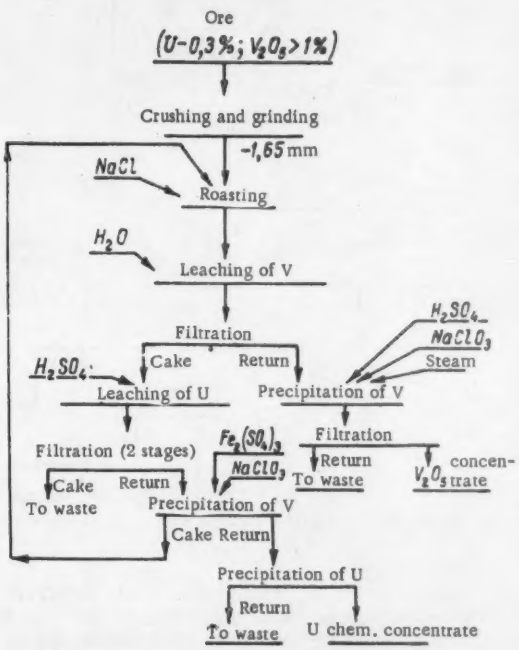

Fig. 3. Flow diagram of complex treatment of uranium-vanadium ores (carnotite ores of the Colorado Plateau, plant at Rifle).

Somewhat different technological schemes are employed at other plants: combined leaching and precipitation of the uranium and vanadium together with their subsequent separation by reducing fusion and leaching of the vanadium from the fusion product, and so forth. In connection with the introduction of adsorption- and solvent-extraction processes, research work is being carried out on the separation or uranium and vanadium in solutions by the method of selective adsorption or solvent extraction.

Uranium—Copper Ores [10]

These ores are not so important as the previous ones. Copper and other nonferrous metals (cobalt, nickel, lead, and others) are encountered comparatively rarely in industrial amounts in uranium deposits, principally in some hydrothermal deposits of vein type. The most typical example of uranium-copper ores is the Rum Jungle deposit in Australia. Figure 4 shows a flow diagram of the complex treatment of the ores of this deposit.

In this ore the copper occurs in both oxide and sulfide forms. In the leaching of the uranium with sulfuric acid with an oxidizing agent (manganese dioxide), the oxidized copper passes into solution almost completely, while the sulfide does so only to a slight extent (principally due to secondary minerals of the precipitation zone).

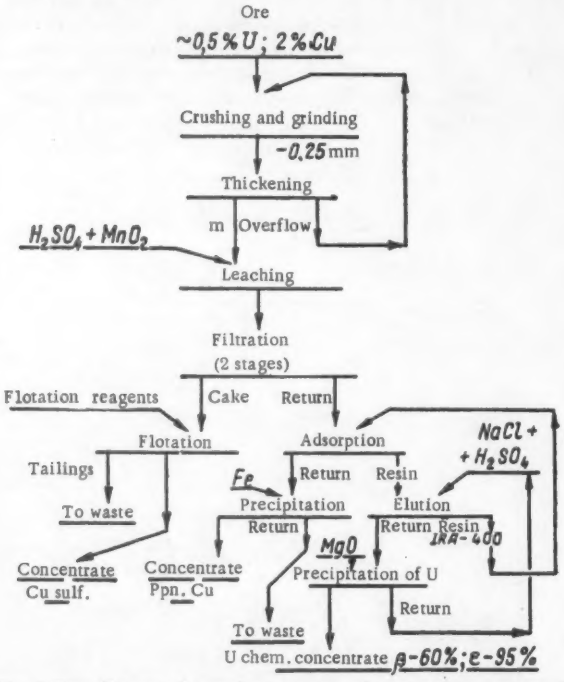

Fig. 4. Flow diagram of complex treatment of uranium-copper ores of the Rum Jungle deposit.

In the adsorption of uranium on ion-exchange resins, the copper, being in the cation form, is not adsorbed, and remains in the waste solutions, from which it is extracted by the usual method of precipitation with scrap iron, giving a concentrate of precipitation copper. The bulk of the sulfide copper remains in the leaching tailings and is extracted from them by flotation, giving a copper sulfide concentrate. Flotation of the copper sulfide minerals after leaching out the uranium ensures the production of a copper concentrate with a waste content of uranium, while in addition the acid treatment of the sulfides, by removing the oxide film from them, promotes their better flotation.

The total amount of copper extracted according to such a scheme is about 90% for very slight expenditure (only the scrap iron and the flotation reagents are consumed, so that the economic extraction of copper is possible with a low copper content of the ore: about 0.2-0.3%). A similar process can be used for the extraction of some other nonferrous metals from uranium ores.

When there is no close intergrowth of uranium minerals with sulfides, the latter can be separated by flotation, which is carried out after the extraction of the coarse grains of uranium pitch by gravitation, before leaching out the uranium. Such a scheme is employed in particular at a factory in Port Radium (Canada).

Uranium-Containing Coal and Shale [11-14]

Uranium-containing coal has no great industrial significance, although isolated deposits, with a content of up to 0.1-0.2% uranium, are known; for example, some kinds of coal from South Dakota (USA). In addition to uranium, this coal often contains molybdenum. The uranium and molybdenum are contained in the coal in the form of organic compounds. The ash content of these coals is usually high (30-60%), and the calorific value low (2000-3000 cal). Coal, and especially shale, are raw materials from which it is difficult to extract the uranium. The uranium can be extracted from them according to two fundamental schemes:

1) Leaching of the uranium with sulfuric acid from raw coal or shale (the tailings being used as fuel);

2) Initial burning of the coal or shale, with extraction of the uranium from the ash.

Leaching of uranium from raw coal is applicable to certain types of coal only. Usually it yields not more than 85-90% of the uranium, and is accompanied by a considerable acid consumption. For most coals, leaching after burning gives a somewhat higher yield, and usually requires less acid than the leaching of the original coal.

The coal must be burnt at a temperature of not more than 600°C, otherwise the yield of uranium in the subsequent leaching of the ash is low. Any unburnt coal can be removed by flotation either before or after leaching. Leaching is effected with sulfuric acid at pH = 0.5 and a temperature of 70-90°C. The acid consumption is 250-500 kg/ton of ash or 75-300 kg/ton of coal. Leaching is best carried out in two stages in countercurrent; the solution produced then has a lower acidity, which facilitates its subsequent treatment. It should also be noted that the pulps obtained are usually difficult to thicken and filter.

Subsequent extraction of the uranium from the solution is possible by adsorption, but the process is made difficult by considerable contamination of the solutions with various impurities, including organic matter and molybdenum. It appears more expedient, therefore, to use solvent extraction; but this is economical only for relatively rich raw material (not less than 0.2% uranium). Triisooctylphosphate and secondary amines are

used for the extraction. The solution subjected to extraction in the investigation of certain kinds of coal in the USA had the following composition (in g/liter):

Uranium	0.72
Divalent iron	8.8
Trivalent iron	6
Sulfate ion	130
Alumina	20

The pH of the solution was 0.9. The following reaction takes place during extraction and re-extraction:

$$4(R_3NH)Cl + UO_2SO_4 + 2SO_4'' \rightleftarrows$$
$$\rightleftarrows (R_3NH)_4UO_2(SO_4)_3 + 4Cl'.$$

Re-extraction of the uranium is carried out with sodium chloride, and the molybdenum is re-extracted with a soda solution and is then precipitated in the form of trisulfide, through the addition of sodium sulfide and sulfuric acid.

In another case, the uranium is extracted from the solution after the sulfuric acid leaching of the ash by means of di-2-ethylhexyl phosphoric acid. If organic compounds are present in the solution, they must first be removed by means of active charcoal. Adsorption of uranium from solutions after leaching of the coal did not give good results, because of the large amounts of organic matter and molybdenum in the solutions. Heat treatment of the coal at 400°C, with the production of a liquid fuel and coke, indicated the possibility of soda leaching the uranium from the coke. Burning the coal dead with soda leaching gives a low yield of uranium and poor filterability. Better results are obtained with a two-stage leaching of the ash with sulfuric acid: in the first (neutral) leaching, iron and molybdenum are precipitated; in the second, the uranium can be extracted by adsorption. The capacity of the IRA-400 resin was up to 68 kg/m^3.

Uranium-containing fuel shale is a more stubborn raw material than coal; the uranium content in it is much lower (0.01-0.03 %). The total reserves of such shale are quite high (Sweden, USA and other countries). To date there has been no industrial exploitation of uranium-containing shale, but in some countries intensive research is going on to discover an economic process for its treatment. For example, in Sweden, an experimental plant is operating on the treatment of uranium-containing shale by the method of acid percolation (with previous screening of the acid-capacious fines and the separation of coarse carbonate inclusions in heavy suspensions). At this plant, phosphate precipitation is used for the extraction of the uranium from the solution: research is likewise being conducted on adsorption and solvent extraction. The cost of uranium extracted from shale is, however, at present about 2-3 times higher than that of uranium extracted from ordinary poor uranium ores (with a uranium content of about 0.1%).

Uranium—Pyritic Ores [15-16]

Pyrite is found in many uranium ores, and can be easily extracted from leaching tailings by sulfide flotation methods and utilized on the spot for the production of sulfuric acid, used for the leaching of uranium. Such a scheme is used in the treatment of gold-uranium ores in South Africa, as well as for some other uranium ores. For example, at a plant at the Lorado Mine (USA) the uranium ore is leached with sulfuric acid with the addition of sodium chlorate as oxidizing agent. After leaching, the pulp undergoes two-stage filtration with washing on revolving filters and continuous removal of the cake; the solution is treated for adsorption of the uranium, and the cake is pulped and is treated for pyrite flotation. At this mine, there are areas where the pyrite content is high but the ore does not contain uranium. These ores are ground separately and treated directly (without leaching the uranium) by flotation, together with the tailings from the uranium ores. In this way a sufficient quantity of pyrite is obtained to supply the Lorado plant with sulfuric acid and to some extent also neighboring plants. The pyrite is roasted in fluidized bed furnaces with a waste-heat boiler plant operating on the waste gases, which assures the heating of the works. Thus, in addition to the utilization of the sulfur contained in the pyrite, the heat produced during its roasting is also utilized.

After acid leaching, the pyrite is treated by flotation very quickly with a low consumption of reagents. For example, at South African plants about 50-100 g of xanthate and about 20 g of pine oil are consumed in the flotation of pyrite for a flotation period of 10-15 min and an electric power consumption of about 6 kw-hr per ton of tailings.

Zirconium—Uranium Ores [17]

Zirconium minerals (zircon, baddelerite and others) often contain uranium. In some cases, a uranium-containing modification of zircon known as malacon is encountered. In the zirconium concentrates of some Brazilian deposits, the uranium content varies from 0.15 to 1.7% (with an average of about 0.5%). In these concentrates the uranium is not represented by independent minerals, but is contained in the crystal lattice of the zirconium minerals, and therefore can be extracted only by the complete decomposition of the latter, which is best accomplished by fusion of the concentrates with alkalis or salts (caustic soda plus sodium fluoride or acid potassium sulfate). In the subsequent leaching of the fusion product, up to 96% of the uranium is separated in the solution. By sulfatization of the concentrates (treatment with sulfuric acid at 300°C), up to 90% of the uranium is separated in the solution. To date, only research work on the extraction of uranium from zirconium concentrates has been published. So far, there is no information concerning the practical application of this process.

Niobium—Tantalum—Uranium Ores [18-19]

Uranium is often contained in niobium-tantalum minerals (the euxenite-polycrase group and others), found in deposits of pegmatite type, as well as in sands. Thus, pyrochlore of the Kaffo (Nigeria) deposit contains 41% niobium pentoxide, 3.6% uranous-uranic oxide, and 3.3% thorium dioxide. Ore of the Newman Island deposit contains 0.53% niobium pentoxide and 0.039% uranous-uranic oxide. The euxenite-columbite concentrate of the Bear Valley (USA) placer deposit, beneficiated on dredgers followed by electromagnetic and electrostatic separation, contains 20-25% niobium pentoxide, 2-5% tantalum pentoxide, 6-10% uranous-uranic oxide, 1-2% thorium dioxide and a total of 18-22% of rare earth oxides. For further complex chemical treatment, the complex is sent to a plant in St. Louis, the planned out-

put of which is about one thousand tons of niobium and tantalum pentoxides annually, with collateral production of uranous-uranic oxide. There is no information on the technology of the treatment of the complex niobium-uranium concentrates.

Uranium—Thorium Ores [20-21]

At the present time, thorium is obtained together with the rare earths exclusively from monazite concentrates, and the scale of its production is quite small (altogether about 400-500 tons annually). Many monazite concentrates contain a certain amount of uranium (about 0.5%), which may be extracted as a collateral recovery during their treatment.

In connection with the use of thorium for the production of nuclear energy, the demand for it will undoubtedly increase, and there is, therefore, an interest in its collateral recovery in the extraction of uranium. Many uranium ores contain complex, uranium-containing minerals of titanium, niobium and tantalum, which also contain thorium. For example, the ores of the very large deposit at Blind River (Canada) contain an average of about 0.1% uranium and about 0.05% thorium dioxide. It is expected that 9000 tons of uranous-uranic oxide will be produced in 1959 from the ores of this deposit, and if collateral recovery is organized, about 4000 to 4500 tons of thorium may be produced in the year, which is about ten times more than the present world production of thorium.

Scientific research work is now being conducted in Canada on the extraction of thorium from uranium ores. Rich uranium-thorium concentrates can be produced from ores by mechanical beneficiation (gravitation and flotation), followed by acid decomposition of the concentrate. Potassium sulfate (about 50 g/liter) is added to the rich uranium-thorium solutions obtained, and the double sulfate of thorium and potassium is precipitated, while the uranium remains in solution and is then extracted in the usual way.

Uranium-Molybdenum Ores [22]

Molybdenum is found in many uranium-containing ores; for example, in uranium-containing coal and shale. The uranium-copper-molybdenum ores of the Lakani deposit in the Philippines contain 0.11% copper, 0.4% molybdenum, 0.03% uranium, etc. Up to the present the molybdenum has not been extracted from these ores, but research work, partly discussed in the section "Uranium-containing coal and shale," is being conducted on their complex exploitation.

Conclusion

It will be seen from the foregoing, that many uranium ores contain a number of valuable accompanying constituents which can be profitably extracted. Many ores of other metals and useful minerals contain a certain amount of uranium, which may also be extracted by collateral recovery. Since in the collateral recovery of uranium or the accompanying constituents, the expenditure involved in the mining, transport, crushing, and grinding of the ores is incurred by reason of the basic constituent, extraction of the accompanying constituents may be profitably realized, even if their content is quite low. The minimum industrial content of the accompanying constituents in the case of side production lies between their maximum permissible content in the dump tailings and the marginal content of the given constituent in the ores.

On this basis, the data given in Table 2, compiled on the basis of the practice of the treatment of ores of complex deposits, will provide an approximate estimate of the possibility of the profitable extraction of accompanying constituents.

Table 2. Approximate Minimum Industrial Content of Accompanying Constituents in Complex Uranium Ores

Type of ore	Minimum industrial content	
	Uranium, in the presence of an industrial content of the other constituents, %	Other accompanying constituents in the presence of an industrial content of uranium
Gold-uranium	0.01	0.5 – 1 g/ton gold
Phosphorus-uranium	0.01 – 0.02	5 – 10% phosphorus pentoxide
Uranium-vanadium	0.02 – 0.03	0.5 – 1% vanadium pentoxide
Uranium-copper	0.02 – 0.03	0.2 – 0.3% copper
Uranium-containing coals and shales	0.03 – 0.05	20 – 30% combustible matter
Uranium-pyritic	0.01 – 0.02	2 – 3% sulfur
Zirconium-uranium	0.01 – 0.02	1 – 2% zirconium dioxide
Niobium-uranium	0.01 – 0.02	0.05 – 0.1% niobium pentoxide
Uranium-thorium	0.01 – 0.02	0.05 – 0.1% thorium dioxide

Remark. All the above-mentioned figures relate to the original unbeneficiated ores, which in most cases can be beneficiated by mechanical methods (gravitation, flotation, etc.).

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Received September 20, 1958.

THE INDUSTRIAL PRODUCTION OF HEAVY WATER

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Heavy water is assuming greater importance in nuclear technology. In reactors it serves as a moderator and coolant simultaneously. Deuterium is also a feasible fuel in nuclear fusion reactions.

This article presents significant data regarding the needs of various countries for heavy water. Brief descriptions are given of heavy water production facilities built at various times, and the output capacities of both active and shut-down installations are given, as are the plans of various countries for development of heavy water production.

Heavy water is an important accessory material in nuclear technology. Its use as a moderator offers definite advantages. In addition it has been demonstrated in recent work in the area of thermonuclear reactions that deuterium obtained from heavy water may in the near future become the material for controlled thermonuclear fusion reactions [1-3], and thus heavy water may become not an accessory but a basic material of nuclear technology. So far, the literature does not contain a complete survey of the present state of industrial production of heavy water abroad. This article is designed to correct this to some extent.

Since 1934 the production of heavy water has grown from the preparation of sample quantities to the production of hundreds of tons annually. At the present time the heavy water needs of such countries as England and the FRG constitute about 100 tons per annum each [4-6].

In view of the rapid development of nuclear power it is not difficult to forsee an enormous increase in heavy water requirements in the near future. The total D_2O needs of the countries of Western Europe that are included in the European Economic Union are estimated at 400-450 tons per annum in the year 1965 [6-8]. The needs of the USA will grow in the near future from several hundred to 1000 tons of heavy water per year [9].

The use of heavy water is limited by its production rate, and most important, by its high cost. Because of this, orders for heavy water, placed primarily in the USA, are relatively low. Thus in 1956 England ordered 50 tons of D_2O, France 30 tons, India 21 tons, Australia 11 tons, Italy 10 tons, Switzerland 7 tons [10-12]; and somewhat later Sweden ordered 26 tons and Norway 16 tons [13]. Until 1955 the sales price of heavy water in the USA was about $180 per kg. In 1955 the Atomic Energy Commission (AEC) of the USA established a new sales price $62 per kg. It is generally assumed that the present sales price is considerably below the actual cost of the heavy water [14-15].

Recently, many topics relating to the production of heavy water have been declassified. A series of surveys of heavy water production methods have been published [15-28]. The most valuable material has been presented in the monograph edited by Murphy [16], and in the surveys by Selak and Finke [20], Augood [21], Becker [24], as well as in the report by Benedict [25].

According to these references, the basic methods of heavy water production are electrolysis of water, rectification of water and hydrogen, chemical exchange between water and hydrogen, and dual temperature exchange between water and hydrogen sulfide. A number of facilities for production of heavy water on an industrial scale by these methods were built, but some of these are at present inactive and on standby, because of their uneconomical characteristics.

The first large facility for production of heavy water was built more than 20 years ago by the private firm "Norsk Hydro" at Rjukan, Norway, based on the plant there for the production of artificial fertilizer using electrolytic hydrogen, which was the largest plant of its type in Northern Europe. The method for production of D_2O was basically the electrolysis of water, and later a combination of electrolysis with chemical exchange between water and hydrogen. Before the Second World War, 1.54 tons of heavy water per year were produced in Norway [24]. During the war a contact system was added to the electrolysis baths, so that isotope exchange reactions between the water vapor and hydrogen could be maintained, and this increased the output to 3 to 4 tons of heavy water per year [15]. At present the rebuilt and expanded facility produces, according to [7,8], 10 to 20, and according to [14-28], 20 to 25, tons per year of D_2O.

Until 1955, the firm "Norsk Hydro" was a basic supplier of heavy water, and its output was completely covered by the demand through the year 1959. The intention was to expand production significantly over the following six years [29], but the decision of the AEC, USA, to sell heavy water at almost one third the price of heavy water at "Norsk Hydro," places this company in a difficult position.

During the years 1942 and 1943 the DuPont Company of the USA erected three plants for the production of D_2O by the water-distillation method at the arsenals located at Morgantown, Newport, and Childersberg [16], [20], [24]. The facilities at Newport and Childersberg were faulty and produced only half their design output. The sum of the outputs of all three plants was about 15 tons of heavy water per year [16-20]. A total of 22.8 tons of heavy water was produced at these plants [24]. Production of heavy water at these plants was halted in October, 1945 because of a decreased demand and because of its high cost.

At about the same time the Canadian "Consolidated Mining and Smelting" firm built a facility for heavy water production by the method of catalytic exchange between hydrogen and steam, together with electrolysis, at its fertilizer plant at Trail, in answer to an order from the USA. The output of the plant was 6 to 8.5 tons of heavy water per year [14], [24]. In 1956 the AEC of the USA shut down this plant as uneconomical. Shortly after, the plant was purchased by Consolidated Mining and Smelting at an undisclosed price [30].

During the Second World War the IG Farbenindustrie in Germany, built experimental installations at Liene for production of heavy water by the method of dual temperature exchange in a water–hydrogen system and in a water–hydrogen sulfide system, while at Bitterfelde it built an installation for rectification of water [6, 17, 25, 31]. However these facilities were destroyed during the war.

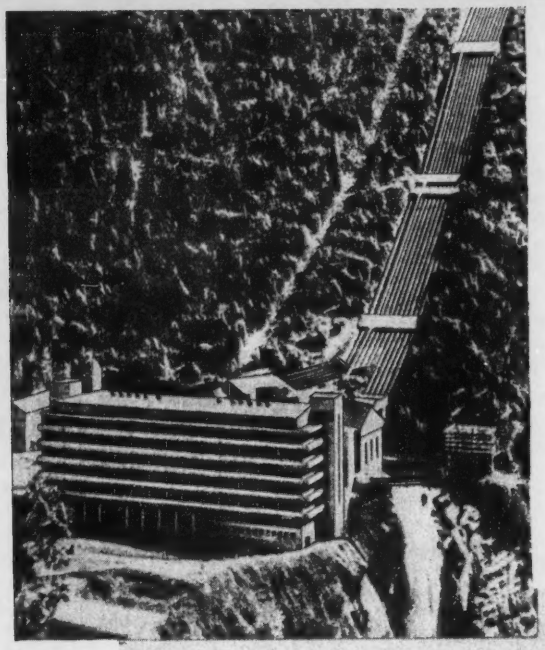
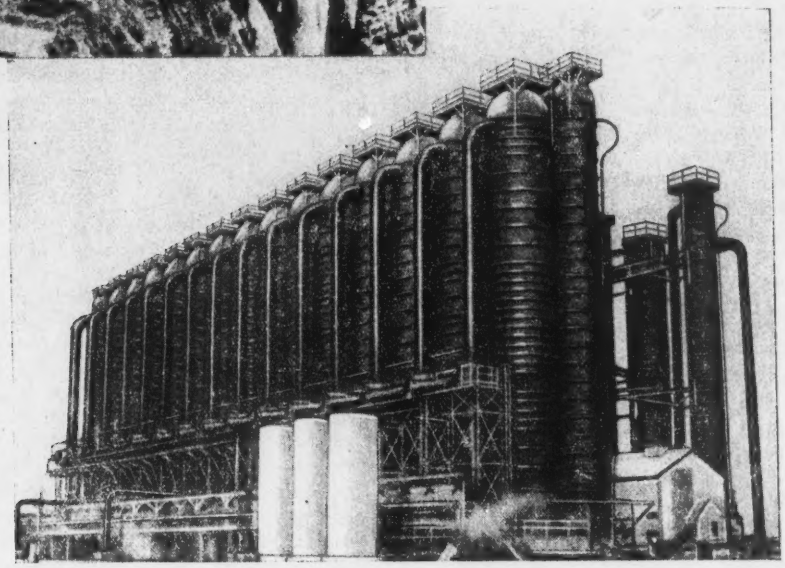


Fig. 1. Heavy water plant of the "Norsk Hydro" Co. at Rjukan, Norway.

Fig. 2. The DuPont heavy water plant at the Wabash River Arsenal at Newport, USA.



During the period 1950 thru 1956 at Savannah River and Dana, USA, the DuPont Company brought into operation two plants for production of heavy water by the method of dual temperature exchange between water and hydrogen sulfide [6-32]. The costs of construction of these plants were, respectively, 164 and 100 million dollars [33]. During 1957 the plant at Dana was shut down and placed on a standby basis [30]. The plant at Dana was placed in a state of standby very carefully, so that in case of necessity it can be placed in operation quickly. The plant at Savannah River continues in operation and produces over 100 tons of heavy water per year [6]. According to Becker's report, [34] which draws information from the report of Babbington and Thayer to the 1957 meeting of the American Society of Chemical Engineers, the total output of both plants is about 800 tons of D_2O per year. The retirement of the plant at Dana to standby status is explained by the fact that the needs of the United States can be satisfied by the plant at Savannah River for the next four years, and that this plant was built somewhat later than the plant at Dana [30]. It was also considered that the cost of heavy water produced at Dana was higher than the cost of the Savannah River product, and that the equipment at Dana was less corrosion resistant [32-35].

Starting apparently in 1954, the firm "Air Liquide," at its nitrogen fertilizer plant at Toulouse, France, has been building an installation for production of heavy water by the method of low temperature hydrogen rectification [5, 24, 36]. The design output of the installation is 2.5 tons of heavy water per year [15-31].

At Kawasaka, Japan, at an ammonium sulfate plant of the Showa Dinco Co., an installation for heavy water production, of 50-kg-per-year output, has been operating since 1943 [22, 37, 38]. It is proposed to link this installation with a powerful electrolytic plant, so that output will reach 10 tons per year and thus satisfy the heavy water needs of Japan [38].

According to many reports [33, 39-43], an installation for production of heavy water by rectification of hydrogen should have gone into operation in the autumn of 1957 at Frankfurt am Main, Federal Republic of Germany. The cost of this installation was 1.7 million dollars [41]. The construction is being conducted by the "Farbwerke Hoechst" with the participation of the Linde Company. The initial output rate is 6 tons per year, but an increase to 10 tons per year is planned [40, 43].

Of the planned but not completed projects, the most interesting are the following two. In 1949-1950 the AEC, USA, assigned the task of developing a design for a heavy water plant using low temperature rectification of hydrogen to the Hydrocarbon Research Co. [15, 20]. This company conducted extensive analysis and experimentation to obtain the essential data. The best location for the plant was chosen at the ammonia plant at Etter, Texas, near which there are enormous sources of cheap natural gas. The design output of the plant is 34.2 tons of heavy water per year. The design was almost complete, but construction was never begun, largely because of the absence of experience in operating such large installations at the temperature of liquid hydrogen, -252°C. At the present time the National Bureau of Standards at Washington has built an experimental installation at Boulder, Colorado, for production of heavy water by the low temperature rectification method [41]. The results of operation of this facility were recently described in detail [44].

In 1954-1955 it was announced [45-47] that England, in cooperation with New Zealand, had begun construction of a heavy water plant on North Island in New Zealand. The heavy water was to be produced by the column distillation of water with "Spraypack" fittings. The sprays were to be used as sources of water and also as sources of energy. Later it became known [48-49] that England had abandoned the construction plans because the water in the sprays had a low deuterium content. In addition, the energy required could not be supplied by the sprays so that capital costs would have been double the initially allotted 17 million dollars [14, 45, 46].

Despite the low price at which the USA is selling heavy water, many countries are trying to establish their own production facilities. This is because the USA places limitations governed by political considerations on the sale of heavy water. In addition, American technicians are required to be present at installations that use heavy water [14-42]. For these reasons India invited the English firm of Costain, John Brown to design a heavy water installation, using hydrogen rectification and producing 12 to 15 tons per year, for the nitrogen fertilizer factory at Nengale [50-53]. The expected cost of the installation is approximately 250 million rupees [54]. Completion of construction was set for 1960 [49]. In addition the possibility of building a heavy water installation at the nitrogen fertilizer plant in Sindri, was examined [52].

The Federal Republic of Germany firm of Pinsh Bamag has completed design of a 100-ton-output plant that will use the catalytic exchange process, already used in Norway and the USA [55-58]. The decision on actual construction of this plant has apparently not been made yet. Schoenemann [59], reports that a plant for heavy water production by the method of dual temperature exchange between water and hydrogen sulfide is also being designed in the FRG, and that this plant will have a 40-ton-per-year output. The proposed cost of the plant is 10 million dollars. As a first step toward mastery of the dual temperature exchange method, Pinsh Bamag are constructing a pilot plant [6], [60]. According to recent information [61-63], it has been decided to build a large test installation for production of heavy water by the dual temperature exchange method, using water and hydrogen, with highly dispersed platinum as a catalyst.

A stock company has been formed in France for the purpose of producing heavy water, and this company, in part, is conducting a study of the dual temperature separation method for a water-hydrogen sulfide system [64, 65]. In Italy, the possibility of producing 3 tons of heavy water per year, using the method of water rectification with geothermic steam, has been studied for location in the province of Tuscany [7, 14, 66]. In Portugal organization of an output of 2 to 3 tons of heavy water per year is expected [7]. In Switzerland it was first expected that two plants would be built, each plant of 3.5 tons-per-year output [14], but the contracts for construction were later cancelled. At present the plans are for construction of a 1.5 ton-per-year facility, based on the operating facility of "Cuna" [8, 26, 67]. In Sweden the firm "Asia" intends to build a facility for production of 20 tons per year [68]. The assumed cost of erecting such a facility is 4 to 6 million dollars. It is expected that the firm "AB Atom Energy" will participate in the construction. A facility for heavy water production by the isotope-exchange method with the utilization of electrolytic hydrogen is being outfitted in Spain [7, 69]. The output of the facility is 1 to 1.5 tons of D_2O per year.

It is interesting to know that the Organization of European Economic Cooperation at the present time is reexamining its initial decision to organize a decentralized system of heavy water production with the aid of private firms [8]. The possibility of outfitting an all-European plant for an output of 500 tons of heavy water per year using the dual temperature exchange method between water and hydrogen sulfide is being considered for a location in Iceland [70]. The cost of construction will be about 200 million dollars; the cost of the heavy water produced here will be at the same level as the present price of American heavy water. England is also prepared to participate in the construction of this plant. It is also considered possible, that for political and economic reasons, the USA will participate in the project.

The table below shows figures on the changing volume of heavy water output from 1935 to 1960, and the relative contributions of the various production methods to the total production. The data presented in the table are, of course, quite approximate, and do not pretend to complete accuracy. The figures given for 1960 are based on operation of all facilities planned at present. It is seen that the greatest quantity of heavy water is produced at present by the dual temperature exchange method in the water-hydrogen sulfide system. This, of course, does not mean that the other methods have no chance for future growth at the expense of this presently predominant method. In particular there is great hope for the catalytic exchange method using hydrogen and ammonia without extraction of nitrogen from the nitrogen-hydrogen mixture. This method may find wide

Changes in the Volume of Heavy Water Production and the Relative Contributions of the Various Methods

Method	Years											
	1935		1940		1945		1950		1955		1960	
	production output											
	tons/year	%	tons/year	%	tons/year	%	tons/year	%	tons/year	%	tons/year	%
Electrolysis	2	100	1.5	100	—	—	—	—	—	—	10	2.0
Electrolysis combined with catalytic exchange	—	—	—	—	6	28.6	28.5	100	28.5	8.7	21.5	4.2
Rectification of water	—	—	—	—	15	71.4	—	—	—	—	1.5	0.3
Rectification of hydrogen	—	—	—	—	—	—	—	—	—	—	27.5	5.4
Dual temperature exchange in a water-hydrogen sulfide system	—	—	—	—	—	—	—	—	~300	91.3	~450	88.1
Total quantity	2		1.5		21		28.5		328.5		510.5	

use in small countries, since it does not require large capital expenditures [71].

Research work in extraction of heavy water has expanded significantly in the past few years. New methods for production are being developed and old methods are being studied for possible improvement. In particular the possibility of using natural raw materials of high deuterium content is being explored [72], as well as the possibility of using mother liquors from the production of salt [73]. New methods for extraction of heavy water on the basis of ice-liquid water equilibrium [74, 75], of absorption fractionating [76], and of biological methods of concentration [77, 78], are being developed. At the symposium on isotope separation held in Amsterdam in April, 1957, considerable attention was devoted to methods of separation of heavy water [79, 80].

The basic problem of research is development of new methods and perfection of old methods of production, so that the cost of heavy water may be significantly lowered. The opinion of one of the leading American investigators, Murphy [16], is that the cost of heavy water will have a great effect on the development of nuclear power.

The development of heavy water separation methods must lead to the perfection of more economical methods of producing heavy water, and the price must then drop. This, in turn, will lead to wider use of heavy water in nuclear power.

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Received May 4, 1958

THE MODERATION LENGTH OF NEUTRONS

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The neutron moderation length is an important constant without which is is impossible to deal with the design of nuclear reactors in all their aspects. A knowledge of the moderation lengths is especially necessary for the determination of the space distribution of the neutrons in a reactor, and for the calculation of the energy spectrum of the moderated neutrons.

In the present work is given an approximate solution of the integral equations satisfied by the space moments of the neutron distribution function in an infinite medium with an infinite, plane, isotropic source. The energy-angle moments of the neutron scattering function are expressed in terms of experimentally determined angular distributions of neutrons of various energies in the case of anisotropic elastic scattering from nuclei. By using experimental results for the total cross section and for the angular distribution in the case of elastic scattering from the nuclei H^1, D^2, Be^9, C^{12}, O^{16} the neutron moderation lengths were calculated for the moderators: water, heavy water, graphite, beryllium, and beryllium oxide.

The results of the calculation were found to be in satisfactory agreement with experimental values.

1. Space Moments of the Neutron Distribution Function

The space-angle moments of the neutron distribution function in an infinite medium with a plane source at $z = 0$,

$$M_l^n(u) = \int_{-\infty}^{+\infty} z^n dz \int d\vec{\Omega} P_l(\mu)\,\psi(z, u, \mu), \tag{1}$$

satisfy the following system of integral equations (see [1]):

$$M_l^n(u) = S(u)\delta_l^0\delta_n^0 + \frac{n}{2l+1}\left[(l+1)M_{l+1}^{n-1}(u) + lM_{l-1}^{n-1}(u)\right] + \int_{-\infty}^{u} M_l^n(u')\frac{\Sigma_s(u')}{\Sigma(u')} f_l(u', u-u')\,du', \tag{2}$$

with u the lethargy; $P_l(\mu)$ the Legendre polynomial; $\psi(z, u, \mu)$ the collision density function;

$$f_l(u', u-u') = \int f(\mu_0, u', u-u') P_l(\mu_0)\, d\vec{\Omega}_0;$$

$f(\mu_0, u', u-u')$ the neutron scattering function; $S(u)$ the spectrum of the neutron source; μ the cosine of the angle between the direction of the velocity and the z axis; μ_0 the cosine of the scattering angle.

We shall be interested in the neutron moderation length $L_s(u)$, which is related to the second space moment by the formula

$$L_s^2(u) = \frac{M_0^2(u)}{2M_0^0(u)}.$$

The system of equations (2) yields the system of three integral equations from which the function $M_0^2(u)$ can be determined:

$$M_0^0(u) = S(u) + \int_{-\infty}^{u} M_0^0(u')\frac{\Sigma_s(u')}{\Sigma(u')} f_0(u', u-u')\,du', \tag{3a}$$

$$M_1^1(u) = \frac{1}{3}\lambda(u)M_0^0(u) + \int_{-\infty}^{u} M_1^1(u')\frac{\Sigma_s(u')}{\Sigma(u')} f_1(u', u-u')\,du', \tag{3b}$$

$$M_0^2(u) = 2\lambda(u)M_1^1(u) + \int_{-\infty}^{u} M_0^2(u')\frac{\Sigma_s(u')}{\Sigma(u')} f_0(u', u-u')\,du', \tag{3c}$$

where $M_0^0(u) = \int_{-\infty}^{+\infty} dz \int d\vec{\Omega}\psi(z, u, \mu)$ is the integrated spectrum of the neutron collisions in the medium and

$$\frac{M_0^2(u)}{M_0^0(u)} = \frac{\int_{-\infty}^{+\infty} z^2 dz \int d\vec{\Omega}\psi(z, u, \mu)}{\int_{-\infty}^{+\infty} dz \int d\vec{\Omega}\psi(z, u, \mu)} = \langle z^2 \rangle$$

is the mean square discharge of neutrons of lethargy u from the source.

If the medium consists of a mixture of different nuclei with scattering cross sections $\Sigma_{si}(u)$, then

$$f(\mu_0, u', u-u') = \sum_i \frac{\Sigma_{si}(u')}{\Sigma_s(u')} f_i(\mu_0, u', u-u'), \tag{4}$$

where $f_i(\mu_0, u', u-u')$ is the distribution function for the ith nucleus. The energy-angle moments of the distribution function

$$\chi_{ln}(u') = \int P_l(\mu)\, d\vec{\Omega} \int (u-u')^n f(\mu, u', u-u')\,du \tag{5}$$

are now introduced. In particular $\chi_{00} = 1$ (on account of the normalization of the function f),

$$\chi_{01}(u') = \int (u-u') f_0(u', u-u')\,du = \xi(u')$$

is the mean logarithmic loss of energy, and

$$\chi_{10}(u') = \int \mu\, d\vec{\Omega} \cdot \int f(\mu, u', u-u')\,du = \overline{\cos\vartheta}(u')$$

is the mean of the cosine of the scattering angle. For a mixture of different nuclei

$$\chi_{ln}(u') = \sum_i \frac{\Sigma_{si}(u')}{\Sigma_s(u')} \chi_{ln}^i(u'), \tag{6}$$

as follows from (4).

Since the angular distributions of elastically scattered neutrons

$$\varphi(\mu_0, u') = \int f(\mu_0, u', u-u')\,du \tag{7}$$

is usually measured independently of their energy, and since for elastic scattering the scattering angle and the loss of energy are related by the formula

$$u - u' = 2\ln\frac{-\mu_0 + \sqrt{\mu_0^2 + M^2 - 1}}{M-1} = G(\mu_0) \tag{8}$$

(M is the ratio of the mass of the nucleus to the neutron mass), it is useful to introduce formulae that express

χ_{ln} directly in terms of the angular distribution. Using the expression (8), we obtain from (5)

$$\chi_{ln}(u') = \int P_l(\mu) G^n(\mu) \varphi(\mu, u')\, d\vec{\Omega}. \qquad (9)$$

For moderate neutron energies the scattering is isotropic in the center-of-mass coordinate system. In this case

$$\varphi(\mu_0, u') = \varphi(\mu_0) = \frac{1}{4\pi M} \frac{\left(\mu_0 + \sqrt{\mu_0^2 + M^2 - 1}\right)^2}{\sqrt{\mu_0^2 + M^2 - 1}},$$

so that

$$\chi_{00} = 1,\ \chi_{01} = 1 - \alpha q_M e^{-q_M},$$
$$\chi_{02} = 2\chi_{01} - \alpha q_M^2 e^{-q_M},$$
$$\chi_{10} = \frac{2}{3M},$$
$$\chi_{11} = \frac{1}{3}\left\{-2M + \frac{10}{3M} + \frac{q_M}{2}\left[M^2 - 3 + \frac{2}{M}\right]\right\},$$

where

$$q_M = 2 \ln \frac{M+1}{M-1},\quad \alpha = \frac{(M+1)^2}{4M}.$$

The scattering of neutrons by protons is isotropic for practically the whole energy range in which we are interested (1 ev $< E <$ 10 Mev), so that for hydrogen ($M = 1$) we obtain

$$\chi_{00} = 1,\ \chi_{01} = 1,\ \chi_{02} = 2,\ \chi_{03} = 6,$$
$$\chi_{10} = \frac{2}{3},\ \chi_{11} = \frac{4}{9},\ \chi_{12} = \frac{16}{27} \cdots.$$

The angular distribution of neutrons scattered by deuterium ($M = 2$) in the center-of-mass coordinate system can be given in the form

$$\varphi_c(\nu, u) = \frac{1 + \alpha(u)\cdot\nu + \beta(u)\cdot\nu^2}{4\pi\left[1 + \frac{\beta(u)}{3}\right]},$$

where ν is the cosine of the scattering angle in the center-of-mass system, and α and β are coefficients that have been determined experimentally in [2].

Using the known relation between the scattering angles in the laboratory and the center-of-mass coordinate systems

$$\nu = \frac{\mu_0\sqrt{\mu_0^2 + M^2 - 1} + \mu_0^2 - 1}{M}$$

and the relation

$$\varphi(\mu_0, u') = \varphi_c[\nu(\mu_0), u']\frac{d\nu}{d\mu_0},$$

we obtain

$$\chi_{00} = 1,\ \chi_{01}(u) =$$
$$= \frac{1}{3+\beta(u)}[2{,}18 - 0{,}948\,\alpha(u) + 0{,}851\,\beta(u)],$$
$$\chi_{02}(u) = \frac{1}{3+\beta(u)}[2{,}54 - 1{,}63\,\alpha(u) + 1{,}32\,\beta(u)],$$
$$\chi_{10}(u) = \frac{1}{3+\beta(u)}[1 + 0{,}85\,\alpha(u) + 0{,}228\,\beta(u)],$$
$$\chi_{11}(u) =$$
$$= \frac{1}{3+\beta(u)}[-0{,}136 + 0{,}526\,\alpha(u) - 0{,}348\,\beta(u)].$$

For sufficiently heavy nuclei ($M \gg 1$) the function $G(\mu)$ can be written in the form of a power series in $\frac{1}{M}$:

$$G(\mu) \simeq \frac{2}{M}(1-\mu) + \frac{(1-\mu^2)(2+\mu)}{3M^2} + \cdots$$

Disregarding all powers of $\frac{1}{M}$ higher than the first, we obtain for $\chi_{ln}(u)$ the expression

$$\chi_{ln}(u) \simeq \left(\frac{2}{M}\right)^n \int (1-\mu)^n P_l(\mu)\varphi(\mu, u)\, d\vec{\Omega}.$$

In particular, $\chi_{00} = 1$, $\chi_{01}(u) = \frac{2}{M}[1 - \varphi_1(u)]$,

$$\chi_{02}(u) = \frac{16}{3M^2}\left[1 - \frac{3}{2}\varphi_1(u) + \frac{1}{2}\varphi_2(u)\right],$$
$$\chi_{10}(u) = \varphi_1(u),$$
$$\chi_{11}(u) = -\frac{2}{3M}[1 - 3\varphi_1(u) + 2\varphi_2(u)],$$

where

$$\varphi_l(u) = \int \varphi(\mu, u) P_l(\mu)\, d\mu$$

can be calculated directly from experimental results for the angular distribution.

In Figs. 1-4 the functions $\chi_{ln}(E)$ for H_2O, D_2O, BeO, C, Be, calculated for the angular distribution given in [2] and [3], are shown.

In the energy ranges where no experimental or calculated results are available, extrapolated values of the moments $\chi_{ln}(u)$ have been used.

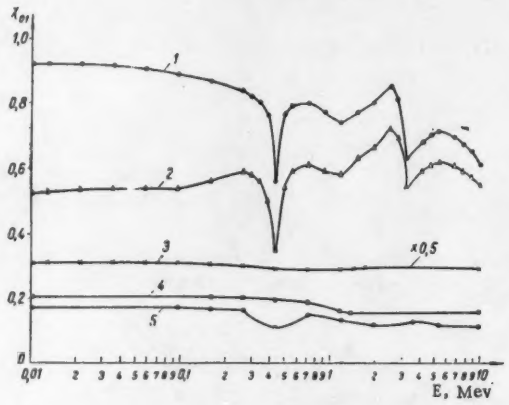

Fig. 1. The function $\chi_{01}(E)$:
1—H_2O; 2—D_2O; 3—C; 4—Be, 5—BeO.

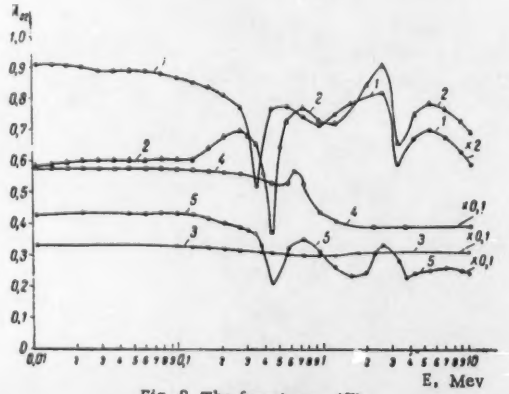

Fig. 2. The function $\chi_{02}(E)$:
1—H_2O; 2—D_2O; 3—C; 4—Be; 5—BeO.

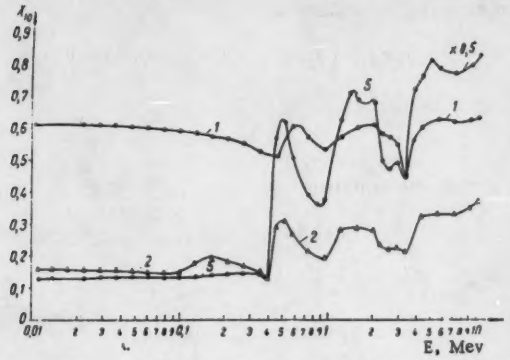

Fig. 3. The function $\chi_{10}(E)$:
1—H_2O; 2—D_2O; 3—BeO.

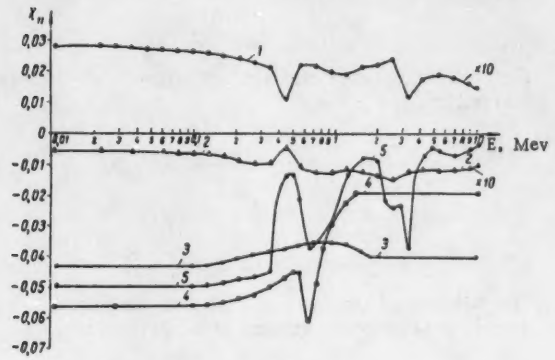

Fig. 4. The function $\chi_{11}(E)$:
1—H_2O; 2—D_2O; 3—C; 4—Be; 5—Be(').

2. Solution of the Moment Equations

For the solution of the system of equations (3) Green's functions are used. Since the kernels of equations (3a) and (3c) are identical, the corresponding Green's functions $Y_0(u, u_0)$ are also the same.

We consider a nonabsorbent medium containing a mixture of nuclei of hydrogen ($M = 1$) and a sufficiently heavy element ($M \gg 1$). Then the equation for the function $Y_0(u, u_0)$

$$Y_0(u, u_0) = \delta(u-u_0) + \int_{u_0}^{u} Y_0(u', u_0)\, c_H(u') \exp(-(u-u'))\, du' + \int_{u-q_M}^{u} Y_0(u', u_0)\, c_M(u')\, f_0^M(u', u-u')\, du', \qquad (10)$$

where $c_H = \frac{\Sigma_H}{\Sigma}$, $c_M = \frac{\Sigma_M}{\Sigma}$, and f_0^M are the cross section and the distribution function of the nucleus of mass $M \gg 1$.

It is known that for $u - u_0 \geqslant 2q_M$ the Fermi neutron spectrum $\frac{1}{\xi(u)}$ is defined. The function $Y_0(u, u_0)$ has moreover, a singularity at $u = u_0$ of the type $\delta(u-u_0)$, which is connected with the first collision of the neutrons. The solution of equation (10) can therefore be written in the form

$$Y_0(u, u_0) = \delta(u-u_0) + y(u, u_0) + \frac{1}{\xi(u)}, \qquad (11)$$

where the function $y(u, u_0)$ represents the difference between the true spectrum and $\frac{1}{\xi(u)}$ for u near to u_0. Since the function $y(u, u_0)$ decreases rapidly for increasing $u - u_0$, then in place of (11) we can write approximately

$$Y_0(u, u_0) \approx \gamma(u_0)\, \delta(u \quad u_0) + \frac{1}{\xi(u)}. \qquad (12)$$

Here $\gamma(u_0)$ is chosen so that the integral $\int_{u_0}^{u} Y_0(u, u_0)\, du$ giving the total number of collisions in the interval (u_0, u) agrees with the correct value for sufficiently large u.

In a small interval of values of u in which the function $y(u, u_0)$ is different from zero, all the cross sections may be considered to be constant and equal to their values for $u = u_0$. Equation (10) then takes the form

$$Y_0(u-u_0) = \delta(u-u_0) + c_H(u_0) \int_{u_0}^{u} Y_0(u'-u_0) \exp(-(u-u'))\, du' + c_M(u_0) \int_{u-q_M}^{u} Y_0(u'-u_0)\, f_0^M(u_0, u-u')\, du'.$$

Taking the Laplace transform of this relation, we obtain

$$X_0(\eta) = 1 + X_0(\eta)\, F_0(\eta, u_0) \quad \text{or} \quad X_0(\eta) = \frac{1}{1 - F_0(\eta, u_0)},$$

where

$$X_0(\eta) = \int_{u_0}^{\infty} Y_0(u'-u_0) \exp(-\eta(u'-u_0))\, du',$$

$$F_0(\eta, u_0) = \int_{0}^{\infty} f_0(u_0, u) \exp(-\eta u)\, du.$$

Here the integral $\int_{u_0}^{u} Y_0(u'-u_0)\, du'$ transforms into $\frac{X_0(\eta)}{\eta} = \frac{1}{\eta[1 - F_0(\eta, u_0)]}$. The function $\frac{X_0(\eta)}{(\eta)}$ has a second order pole for $\eta = 0$, since $F_0(0, u_0) = \int_{0}^{\infty} f_0(u_0, u)\, du = 1$.

Taking residues at the point $\eta = 0$ we find that for large u, the integral we are seeking is given by the expression

$$\int_{u_0}^{u} Y_0(u'-u_0)\, du' \approx \frac{u-u_0}{-F'(0, u_0)} + \frac{1}{2} \frac{F''(0, u_0)}{[F'(0, u_0)]^2},$$

where

$$F'(0, u_0) = -\int_{0}^{\infty} u f_0(u_0, u)\, du = -\chi_{01}(u_0) = -\xi(u_0),$$

$$F''(0, u_0) = \int_{0}^{\infty} u^2 f_0(u_0, u)\, du = \chi_{02}(u_0).$$

This integral can also be obtained from the approximate expression (12) in the form

$$\int_{u_0}^{u} Y_0(u'-u_0)\, du' = \gamma(u_0) + \frac{u-u_0}{\xi},$$

whence

$$\gamma(u_0) = \frac{1}{2} \frac{\chi_{02}(u_0)}{\chi_{01}^2(u_0)}. \qquad (13)$$

In a similar way we obtain an approximate expression for the Green's function $Y_1(u, u_0)$ for equation (3b). It satisfies the equation

$$Y_1(u, u_0) = \delta(u-u_0) + \int_{u_0}^{u} Y_1(u', u_0) c_H(u') \exp(-3/2(u-u'))\, du' + \int_{u-q_M}^{u} Y_1(u', u_0) c_M(u') f_1^M(u', u-u')\, du'. \quad (14)$$

First of all, we find the expression $Y_1(u, u_0)$ for $u-u_0 \gg q_M$. Noting that when $u-u_0 \gg q_M$ $Y_1(u', u_0) c_M(u')$ and $f_1^M(u', u-u')$ do not vary greatly in the interval $u-q_M < u' < u$, we substitute in the last integral

$$Y_1(u', u_0) c_M(u') \approx Y_1(u, u_0) c_M(u)$$

and

$$f_1^M(u', u-u') \approx f_1^M(u, u-u'),$$

and we obtain the equation

$$Y_1(u, u_0)[1 - c_M(u)\chi_{10}^M(u)] = \delta(u-u_0) + \int_{u_0}^{u} Y_1(u', u_0) c_H(u') \exp(-3/2(u-u'))\, du'.$$

When this equation is multiplied by $\exp(3/2u)$ and then differentiated with respect to u a differential equation of the first order is obtained with the solution

$$Y_1(u, u_0) = \frac{\delta(u-u_0)}{1 - c_M(u_0)\chi_{10}^M(u_0)} + \frac{c_H(u_0) \exp\left(-\frac{3}{2}\int_{u_0}^{u} \frac{1-\chi_{10}(u')}{1-c_M(u')\chi_{10}^M(u')}\, du'\right)}{[1-c_M(u_0)\chi_{10}^M(u_0)][1-c_M(u)\chi_{10}^M(u)]}. \quad (15)$$

It is not difficult to verify that the function (15) gives, for constant cross sections, the correct value for $\int_{u_0}^{\infty} Y_1(u, u_0)\, du$.

However, as in the previous case, the function (15) does not take into account the oscillations of $Y_1(u, u_0)$ for $u-u_0 \approx q_M$. In order to take these oscillations into account we add to the solution (15) the function $A(u_0)\delta'(u-u_0)$, defining $A(u_0)$ in such a way that the integral $\int_{u_0}^{\infty} (u-u_0) Y_1(u, u_0)\, du$ will have its correct value. Taking the cross sections as constant and equal to their value for $u=u_0$ in the interval $u-u_0 \approx q_M$, where the correct solution differs from (15), we obtain with the help of the Laplace transform of equation (14) the formula

$$X_1(\eta) = 1 + X_1(\eta) F_1(\eta, u_0)$$

or

$$X_1(\eta) = \frac{1}{1 - F_1(\eta, u_0)}.$$

Since

$$\int_{u_0}^{\infty} (u-u_0) Y_1(u-u_0)\, du = -\left[\frac{\partial}{\partial \eta} X_1(\eta)\right]_{\eta=0} = -\frac{F_1'(0, u_0)}{[1-F_1(0, u_0)]^2},$$

then, using the relation

$$F_1(0, u_0) = \int_0^{\infty} f_1(u_0, u)\, du = \chi_{10}(u_0) = \overline{\cos}\,\vartheta(u_0),$$

$$F_1'(0, u_0) = -\int_0^{\infty} u f_1(u_0, u)\, du = -\chi_{11}(u_0),$$

and also the equality

$$\int_{u_0}^{\infty} \left\{ A(u_0)\delta'(u-u_0) + \frac{\delta(u-u_0)}{1-c_M\chi_{10}^M(u_0)} + \frac{c_H(u_0) \exp\left(-\frac{3}{2}\frac{1-\chi_{10}}{1-c_M\chi_{10}^M}(u-u_0)\right)}{[1-c_M\chi_{10}^M(u_0)][1-c_M\chi_{10}^M(u_0)]} \right\} (u-u_0)\, du = -A(u_0) + \frac{c_H(u_0)}{1-c_M\chi_{10}^M(u_0)} \cdot \frac{4}{9} \frac{(1-c_M\chi_{10}^M(u_0))^2}{(1-\chi_{10})^2},$$

we obtain

$$A(u_0) = -\frac{c_M(u_0)\chi_{11}^M(u_0)}{[1-\chi_{10}(u_0)]^2}.$$

The following approximate expression is therefore arrived at for $Y_1(u, u_0)$:

$$Y_1(u, u_0) = -\frac{c_M(u_0)\chi_{11}^M(u_0)}{[1-\chi_{10}(u_0)]^2}\delta'(u-u_0) + \frac{\delta(u-u_0)}{1-c_M(u_0)\chi_{10}^M(u_0)} + \frac{c_H(u_0) \exp\left(-\frac{3}{2}\int_{u_0}^{u} \frac{1-\chi_{10}}{1-c_M\chi_{10}^M}\, du'\right)}{[1-c_M\chi_{10}^M(u)][1-c_M\chi_{10}^M(u_0)]}. \quad (16)$$

The solution of the system (3) using the Green's functions Y_0 and Y_1 may be written in the form

$$M_0^2(u) = \frac{2}{3}\int_{-\infty}^{u} S(u_0)\, du_0 \int_{u_0}^{u} du' Y_0(u, u')\lambda(u') \times \int_{u_0}^{u'} du'' Y_1(u', u'')\lambda(u'') Y_0(u'', u_0). \quad (17)$$

Substituting the approximate expressions (12) and (16) in (17) we obtain for $L_e^2(u) = \frac{M_0^2(u)}{2M_0^0(u)}$ the formula

$$L_e^2(u) = \frac{\lambda(u)\lambda_{tr}(u)}{3}[\gamma(u) B_1(u) - A(u)] + \int_{-\infty}^{u} \frac{\lambda(u_0)\lambda_{tr}(u_0)}{3}[\gamma(u_0) B_2(u_0, u) - A(u_0)] S(u_0)\, du_0 + \int_{-\infty}^{u} \frac{\lambda(u')\lambda_{tr}(u')}{3\xi(u')} \times B_2(u', u) du' \int_{-\infty}^{u'} S(u'')\, du'', \quad (18)$$

where

$$B_1(u) = \frac{1-\chi_{10}(u)}{1-c_M(u)\chi_{10}^M(u)}\left[1 + \frac{\xi(u)}{\lambda(u)}\int_{-\infty}^{u} c_H(u') \times \frac{\lambda(u') \exp\left(-\frac{3}{2}\int_{u'}^{u} \frac{1-\chi_{10}}{1-c_M\chi^M}\, du''\right)}{\xi(u')[1-c_M\chi_{10}^M(u')]} du'\right],$$

$$B_2(u_0, u) = \frac{1-\chi_{10}(u_0)}{1-c_M(u_0)\chi_{10}^M(u_0)}\left[1 + \frac{c_H(u_0)}{\lambda(u_0)} \times \int_{u_0}^{u} \frac{\lambda(u')}{1-c_M\chi_{10}^M(u')} \cdot \exp\left(-\frac{3}{2}\int_{u_0}^{u'} \frac{1-\chi_{10}}{1-c_M\chi_{10}^M}\, du''\right) du'\right],$$

Comparison of calculated and experimental values of moderation lengths $L_s^2(u)$ in cm²

Source	H_2O	D_2O	Be (ρ=1.85)	BeO (ρ=3.0)	C (ρ=1.6)
Work [4]	30.4± 0.4	100± 5	80± 2	–	311± 3
Formula (18)	29	–	–	–	–
Formula (20)	29.2	106	80	92	318.4
τ (u)	19.5	92.5	74.6	86.8	304.1

$$A(u)=\frac{c_M(u)\,\chi_{11}^M(u)}{2\xi(u)\,[1-\chi_{10}(u)]},$$

$$\lambda_{tr}(u)=\frac{\lambda(u)}{1-\chi_{10}(u)}.$$

The terms which decrease exponentially for increasing u and the expressions $\frac{\partial}{\partial u}\sim\chi_{11}^M(u)$ are not taken into account in (18). The formula (18) is asymptotically accurate (with an error that decreases exponentially) for pure hydrogen and also gives asymptotically accurate values of $L_s^2(u)$ for a mixture of hydrogen with heavy nuclei and for a hydrogen-free medium in the case of constant cross sections.

Actually, substituting

$$c_M=0,\ c_n=1,\ \gamma=1,\ \xi=1,\ \chi_{10}=\frac{2}{3},$$

in (18) we obtain

$$L_s^2(u)=\frac{1}{3}\left\{\lambda^2(u)\left[1+\frac{1}{\lambda(u)}\int_{-\infty}^{u}\lambda(u')\exp\times\right.\right.$$

$$\times\left(-\frac{u-u'}{2}\right)du'\Big]+\int_{-\infty}^{u}\lambda^2(u_0)\left[1+\frac{1}{\lambda(u_0)}\times\right.$$

$$\times\int_{u_0}^{u}\lambda(u')\exp\left(-\frac{u'-u_0}{2}\right)du'\Big]S(u_0)\,du_0+$$

$$+\int_{-\infty}^{u}u')\left[1+\frac{1}{\lambda(u')}\int_{u'}^{u}\lambda(u'')\times\right.$$

$$\left.\left.\times\exp\left(-\frac{u''-u'}{2}\right)du''\right]\cdot\int_{-\infty}^{u'}S(u''')\,du'''\right\},\qquad(19)$$

which coincides with the accurate formula for hydrogen (see [1]). In the other case, if it is assumed that the cross sections do not vary greatly, then on carrying out the integrations in B_1 and B_2 we obtain

$$L_s^2(u)=\frac{\lambda(u)\,\lambda_{tr}(u)}{3}\left[\gamma-\frac{\chi_{11}(u)}{2\xi(1-\chi_{10}(u))}\right]+$$

$$+\int_{-\infty}^{u}\frac{\lambda(u')\,\lambda_{tr}(u')}{3}\left[\gamma-\frac{\chi_{11}(u')}{2\xi(u)\,(1-\chi_{10}(u'))}\right]\times$$

$$\times S(u')du'+\int_{-\infty}^{u}\frac{\lambda(u')\,\lambda_{tr}(u')}{3\xi(u')}\int_{-\infty}^{u'}S(u'')\,du'',\qquad(20)$$

which agrees with the exact formula for constant cross sections (see [1]). Formula (20) can be obtained starting from the following expression for $Y_1(u, u_0)$:

$$Y_1(,u,u_0)=-\frac{\chi_{11}}{(1-\chi_{10})^2}\delta'(u-u_0)+\frac{\delta(u-u_0)}{1-\chi_{10}},\qquad(21)$$

i.e. by dealing with the hydrogen and the heavy nuclei in the same way. The last term in formula (20) corresponds to the age of the neutrons $\tau(u)$ while the first two terms are corrections to the last term to deal with the first and final collisions.

3. Moderation Lengths for H_2O, BeO, D_2O, Be, and C, and Comparison with Experiment

Results of calculations of the neutron moderation length for the indium resonance energy E_{In}=1.46 ev are given in the table. For comparison are given values of ths squares of the moderation lengths $L_s^2(u)=\tau(u)$ calculated for the same data but with different degrees of approximation. As is evident from the table the calculated values agree satisfactorily with the measured values.

It should be mentioned that in the case of ordinary water the main term in L_s^2 yields high energies E>100 kev. In this energy range experimental results for the total cross section, and especially for the angular distribution, are not sufficiently accurate so that it is not possible, for example, to allow reliably for the resonance structure of the cross sections of the O^{16} nucleus, or for correlation between the resonances in the total cross section of O^{16} and the angular distributions. Therefore, more accurate calculations of the moderation length in water are probably not possible at the present time.

In conclusion the authors wish to thank Doctor of Physical and Mathematical Science G. I. Marchuk for useful discussions, and also B. S. Gudkov, Z. P. Drobyshev, and Z. I. Shemetenko for carrying out the calculations.

*Part of the last term of (19) is illegible in the Russian text.

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Received June 21, 1958

SPECTROMETRIC INVESTIGATIONS OF THE γ-EMISSION OF NATURAL EMITTERS

G. M. Voskoboinikov and N. P. Kartashov

An examination is made of the results of a theoretical calculation of the hard γ-ray spectrum of uranium and thorium uniformly distributed in rock and in equilibrium with their decay products. The energy distribution of the pulses produced by this γ-emission in a scintillation γ-spectrometer is calculated. The results are presented in the form of graphs which enable one to carry out a calculation of the spectral effects expected in a spectrometric investigation of natural γ-emission. The necessity for such calculations arises in the development of γ-spectrometric methods of exploration and prospecting for useful minerals, such methods having lately come into wide use in geophysics.

Calculated coefficients of the equations for the separate determination of the uranium and thorium content in rock are also presented. Calculated spectrographs are presented, with a discussion concerning questions of the accuracy of spectrometric investigations when employed under the conditions of field geophysical observations.

Introduction

The application of scintillation counters in geophysics enlarges the possibilities of radiometric investigations by allowing one to measure not only the intensity of natural and artificially produced γ-emission in rock, but also some of its spectral characteristics, thus guaranteeing a more unambiguous interpretation of the results of observations. The necessity for a calculation of the effects expected with the use of a given apparatus and method of measurement arises as a result of the need both for a correct estimate of the possibilities of spectrometric methods and for a rational choice of the parameters of a measuring apparatus intended for the solution of concrete geophysical problems. However, the difficulty of the computational operations involved in the calculation of the spectral composition of the recorded γ-radiation and the energy distribution of the pulses in a scintillation spectrometer arising from this radiation, forces the designer frequently to use the method of empirical selection of the parameters of the apparatus, which does not always lead to correct results.

In order to make the calculations easier, the authors determined the hard (as being the more stable) part of the γ-emission of the basic natural emitters, together with the corresponding spectral distribution of the secondary β-particles arising in phosphors of various compositions and which are directly recorded by the γ-spectrometer.

1. The γ-Spectra of Uranium and Thorium

Figs. 1 (a and b) show the results of a calculation of the spectral distribution of primary and scattered γ-radiation with energy $E_\gamma \geqslant 0.5$ Mev in an infinite homogeneous medium (rock) containing 0.01% uranium and 0.023% thorium* in equilibrium with their decay products. The solid curves represent the differential spectra of the scattered γ-radiation in units of quanta/cm^2 · min · Mev. The number of scattered γ-quanta in a given energy interval is numerically equal to the area enclosed by the curve and the ordinates defining the interval. The intensity of the monochromatic lines of the primary radiation (allowing for attenuation in the rock) is represented in the same units by the areas of the trapezoids drawn with dotted lines above the curve of spectral intensity. The vertical arrows indicate the true energies of the quanta of the spectral lines.†

The γ-emission of the uranium in equilibrium was identified in the calculations as the emission of radium and its decay products. The spectral composition of the radiation was taken from [1], the data of which are also given in a review article [2]. The spectral composition of the γ-emission of thorium in equilibrium was taken from [3], in which the results of various authors were generalized. In the spectral range which was examined, they agree satisfactorily with the data of a more recent generalization [2].

The spectrum of the scattered γ-rays was obtained by an exact solution of the equation of motion [4], which is discussed elsewhere [5] in greater detail. The calculations were made for the case of granite, however, the results are practically the same in the region of the spectrum under consideration for all rocks.

The spectral composition of the radiation over the active semi-infinite region is theoretically different from the composition of the radiation within the active medium, in that it has a different distribution and less relative intensity of scattered rays (due to absence in its composition of so-called back scattered rays). The intensity of the back scattered rays is very low in the hard portion of the spectrum, as is shown by experimental data [6] and general theoretical considerations. Therefore, all further results directly pertaining to measurements in blast holes and drilled holes remain essentially the same for measurements on the surface of the earth, if the absolute value of the intensity is decreased by a factor of two.

It is evident from the graphs which are presented that the intensity of the scattered rays is small in comparison with the intensity of the primary radiation in the range of the spectrum considered. Therefore, all of the characteristics of the spectra of the γ-emission of uranium and thorium are here completely unchanged. Calculations show that in the energy range $E_\gamma < 0.5$ Mev, the primary radiation does not practically separate into a background of intensely scattered rays, the spectral distribution of which does not essentially depend on the type of emitter, but rather on the rock composition. The spectral distribution and methods of using the γ-emission of this energy region for the solution of geophysical problems are discussed elsewhere [3], [7] in detail.

*All calculations were made for concentrations of emitters mutually equivalent in their integral intensities of γ-radiation, so as to obtain a clearer separation of the spectral characteristics of uranium and thorium (see Section 4).

†The area of the third trapezoid in Fig. 1,b represents the total intensity of the fourth line.

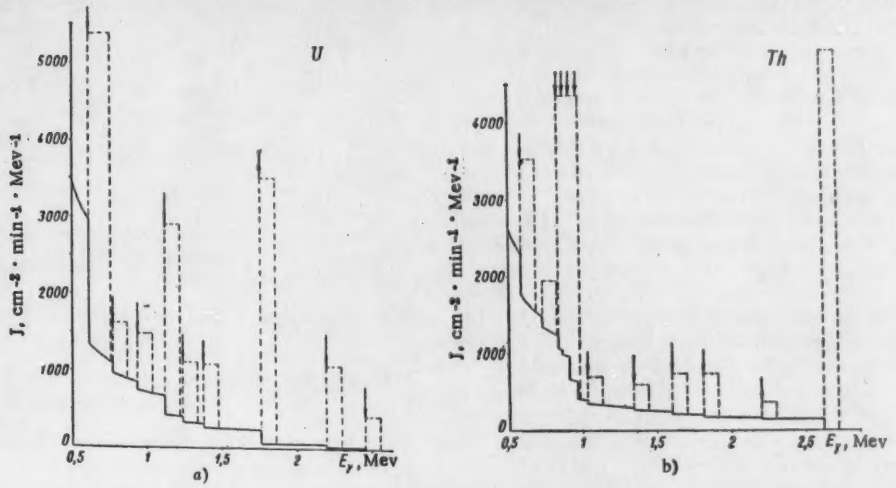

Fig. 1. Spectra of the γ-radiation of uranium (a) and thorium (b) in equilibrium and uniformly distributed in rock.

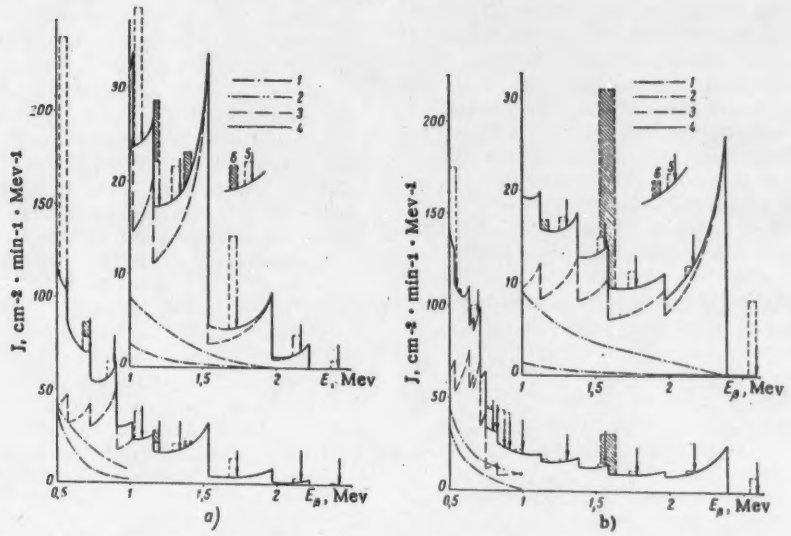

Fig. 2. Spectra of secondary β-rays formed in a NaI(Tl) crystal under the action of the γ-radiation of uranium (a) and (b) in rock.

2. Spectra of Secondary Electrons in the Phosphor

A scintillation counter in conjunction with a linear pulse amplifier gives electrical pulses whose amplitudes are proportional to the energies of the secondary electrons (photoelectrons, Compton recoil electrons and electron-positron pairs) arising in the phosphor of the counter under the action of the γ-rays (see, for example, [8]).

The spectral distribution of the β-particles formed per minute in 1 gram of the NaI(Tl) crystal under the action of the γ-radiation of uranium and thorium is shown in Fig. 2, a and b. The well-known Klein-Nishina formula was used to compute the spectra of the recoil electrons. The values of effective cross sections for photoabsorption and pair formation were taken from [9]. The curves correspond to: 1) photoelectrons from the continuous spectrum of scattered γ-rays, 2) recoil electrons from the continuous γ-ray spectrum, 3) recoil electrons from the γ-rays from monochromatic lines, 4) the total density of distribution of β-particles in the continuous spectrum (the sum of the ordinates of curves 1-3). The uncrosshatched area of the dotted trapezoid 5 represents the number of monochromatic β-particles produced in the process of photoabsorption of the γ-quanta of the primary spectral line; the number of approximately "monochromatic" (with respect to the total particle energy which is effectively used) electron-positron pairs formed by the primary γ-quanta is represented by the area of the crosshatched trapezoid 6. By an accidental coincidence, the positions of the trapezoids on the energy axis added up to their area. The true energy of the particles is indicated by arrows.

On the basis of the graphs presented, one can calculate the spectral distribution of β-particles in phosphors of different composition. Thus, for organic phos-

phors which do not contain heavy elements but which do possess an increased electron density, it follows that one should exclude the photoelectrons and pairs (curves 1 and the dotted trapezoids) from the number of β-particles, while the ordinates of the new total curves should be multiplied by a factor 1.25. In order to obtain the spectrum of the β-particles in a CsI(Tl) crystal, one should multiply the number of photoelectrons and pairs formed in the NaI(Tl) crystal by the factor 1.28, and make a new summation of them with the recoil electrons. The spectra of the secondary particles in the KI(Tl) and NaI(Tl) crystals are practically identical.

The intensity of the radiation recorded by the scintillation spectrometer can be computed for an arbitrary region of the spectrum by graphical integration of the differential spectra of the β-particles (presented in Figs. 2, a and b). To get absolute values of the intensity appropriate for an actual spectrometer and a given content of emitter in the rock, the result of the integration is multiplied by the weight of the scintillator in grams and by the ratio of the calculated content of emitter to the content used in constructing the graphs (see § 1).

Edge effects at the boundaries of the scintillator and multiple interactions of the γ-quanta with the phosphor material were not considered in the calculation of the β-spectra. Consequently, with the aid of the diagrams which have been discussed, accurate results can only be obtained for scintillators of isometric form and moderate dimensions (approximate weight 50-100g). The accuracy is reduced for thin scintillators comparable in thickness with the mean free path of β-particles, as it is for very large scintillators for which the double interaction of the γ-quanta becomes important.

3. Effects Arising from the Apparatus

The pulse-height distribution recorded by the scintillation γ-spectrometer depends on the band width excised from the secondary β-ray spectrum in the process of recording (the width of the "window" of the spectrometer). By decreasing the width of the window, it is theoretically possible to increase without limit the relative intensity of a purely monochromatic line of a photoeffect in the K-shell in comparison with the corresponding portion of the continuous β-spectrum. The resolving power of the apparatus is thus increased, and the form of the recorded spectra becomes closer to the form of the spectrum of the γ-radiation being investigated, being amplified in the region of low quantum energies on account of the greater photoabsorption of soft radiation in the phosphor.

Actually, the resolving power is limited by the potentialities of the γ-spectrometers, whose effective window width cannot be made less than a certain extent. Thus, in one experiment [10], the nominal window width was 0.02-0.05 Mev; however, the effective width of window (determined by the half-width of the photopeak) was about 0.1 Mev (in units of energy of the β-particles). For recording low levels of radiation (usual in geophysical investigations), the window width should be further increased to obtain a reliable pulse-counting rate. The photoeffect is masked by the stronger recoil electron current when the window width is increased, and the picture of the pulse energy distribution is radically smoothed out. As an example, Fig. 3 shows the form of the spectra which the radiation of uranium and thorium must have for a window width of 0.25 Mev.

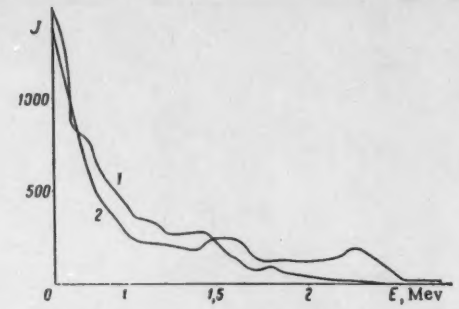

Fig. 3. Calculated γ-spectra for a scintillation spectrometer with a NaI(Tl) crystal for a window width of 0.25 Mev.

1) Emission from uranium in equilibrium in rock; 2) emission from thorium in equilibrium in rock.

The spectra were obtained by integration of the β-spectra between limits which were movable along the energy axis in such a way that their difference always was 0.25 Mev. The number of pulses counted per minute by the spectrometer with a NaI(Tl) crystal (weighing 50 g) was plotted on the ordinate, and the position of the center of the window on an energy scale was plotted on the abscissa. Curve 1 is for measurements in a blast hole with the surrounding rock containing 0.01% uranium, and curve 2 is for a 0.023% content of thorium.

By comparing the graphs in Figs. 1 and 3, one can see that the similarity between the initial γ-spectra and the spectra recorded using a relatively wide window vanishes almost completely; the original form of the composite γ-spectrum cannot be recovered in the form of the spectrum. This fact must be especially carefully considered in experimental investigations of the spectral composition of γ-emission in connection with the solution of geophysical problems (investigations of the spectra of γ-rays scattered in rock, investigations of the spectral composition of induced γ-emission in the activation method, etc.). In particular, the effect described must explain the "washed-out" appearance of the spectra of the scattered radiation of radium and mesothorium which had been obtained experimentally [10].

4. Optimum Conditions and Accuracy of the Separate Determination of Uranium and Thorium Content in Rock

It is well known [11] that the contents of uranium and thorium in equilibrium in rock can be separately determined (if it is possible to neglect the emission from potassium) from a system of two equations of the following type:

$$a_1 U + b_1 Th = J_1, \quad a_2 U + b_2 Th = J_2, \qquad (1)$$

where J_1 and J_2 are the gamma intensities measured by the scintillation spectrometer for two essentially different levels of pulse discrimination; U, Th are the contents of uranium and thorium in per cent; $a_{1,2}$ $b_{1,2}$ are the respective radiation intensities recorded in the rock with unit content of uranium and thorium for the first and second levels of discrimination. We will refer to the ratio of the coefficients $\xi = \frac{a}{b}$ as the thorium equivalent for uranium for the γ-rays of the given portion of the spectrum.

The accuracy of the determination of the unknowns (U and Th) from the system of equations (1) depends on two factors: 1) on the stability of the solution (which

is greater, the greater the relative change in the thorium equivalent of uranium for a transition from the first discrimination level to the second); 2) on the relative accuracy of the free terms, which increases, other conditions remaining the same, proportionally to the square root of their values.

Let us agree to express a, b, and J in the number of pulses recorded per minute by the given apparatus. Then, neglecting the error in determining the natural background, we can find the relative mean square error in the measurement of the free term according to the formula

$$\frac{\Delta J}{J} = \frac{1}{\sqrt{Jt}}, \qquad (2)$$

where t is the time of the measurement in minutes. In addition, the relative squared error in the determination of the uranium content is given by

$$m_U = \frac{\Delta U}{U} = \frac{1}{\sqrt{Ut}} \cdot \frac{\sqrt{b_2^2 A_1 + b_1^2 A_2}}{b_2 A_1 - b_1 A_2}, \qquad (3)$$

where $A_i = a_i + q b_i$; $q = \frac{Th}{U}$ is the ratio of the thorium and uranium contents in the rock; ΔU is the absolute average squared error in the determination of the uranium (in per cent of the mass of rock). The table lists values of a, b and ξ, corresponding to measurements (made with a scintillation spectrometer using a 50-g NaI(Tl) crystal) for various discrimination levels. The values of a and b for zero discrimination depend critically on the noise level of the apparatus (conditionally taken for the zero of discrimination) and the composition of the rock being investigated, and therefore are determined approximately. The only other things of importance to us are their orders of magnitude and the value of the thorium equivalent ξ.

Discrimination level, Mev		a	b	ξ	m_U,%
Lower	Higher				
0		$2.3 \cdot 10^3$	10^3	2.3	–
0.5	–	$285 \cdot 10^2$	$125 \cdot 10^2$	2.28	244
1.0	–	$821 \cdot 10^2$	$460 \cdot 10^2$	1.78	18.7
1.5	–	$209 \cdot 10^2$	$284 \cdot 10^2$	0.74	6.8
2.0	–	$23 \cdot 10^2$	$127 \cdot 10^2$	0.18	6.7
2.5	–	0	$9 \cdot 10^2$	0	22.0
1.0	1.5	$612 \cdot 10^2$	$176 \cdot 10^2$	3.48	–
2.0	2.5	$23 \cdot 10^2$	$118 \cdot 10^2$	0.195	7.0

The value of ξ found in measuring the total γ-intensity depends only slightly on the type of indicator, and its value can be taken as 2.3 to a sufficient degree of accuracy; these facts are known from many experimental investigations and theoretical calculations [3]. The values of the coefficients for other discrimination levels are obtained exactly by integration of the β-spectra considered above.

In practice, the separate content of emitters is frequently determined by observation of the radiation intensity for null discrimination (the first equation of system (1)) and for threshold discrimination at one of the higher levels (second equation). The values of m_U corresponding to this system for various discrimination levels are listed in lines 2-6 of the table. The last two lines give the coefficients of the equations and the value m_U for a system which uses the differential intensity; this system is discussed in more detail below.

The values of m_U were calculated according to formula (3) for t=1; U = 0.01% and q = 2.3. If we take the statistical weight of such a measurement as unity, the weight of every other measurement for constant q can be found by use of the formula

$$p = 100 U t. \qquad (4)$$

The dependence of p on q has a more complicated form and can be found with the aid of formula (3) as the square root of the inverse ratio of their respective errors.

It is clear from the table and the formulas listed above that the stability of the solution increases for a system with a certain threshold discrimination when its level is raised, while the value of the second free term and the relative accuracy in its determination decrease. The increase in stability of the solution is a more decisive factor at low levels of discrimination, and the error in the determination of the unknowns decreases; at high levels, the increasing relative error of the free term acquires decisive importance, and as a result, the error in the determination of the unknowns again increases. The optimum discrimination level corresponding to the least value of m_U is between 1.5 and 2 Mev for the indicator and emitters considered.

By combining the differences of the coefficients listed in the table by pairs in a different manner, one can obtain all of the systems of equations of the type of equation (1) for various cases of measuring the differential radiation intensity for window widths of 0.5 Mev and above. Out of a number of these systems, one system possesses the greatest stability of solution; the coefficients of itsequations and the value of m_U which corresponds to it are listed in the table.

By an analysis of the graphs given in Fig. 2, it is easy to convince oneself that this system is constructed according to the principle of maximum separation of the recorded radiation intensity into "thorium" (the second equation of the system, corresponding to measurement of the radiation in the spectral range 2-2.5 Mev) and "uranium" (first equation, spectral range 1-1.5 Mev).

The thorium content is determined from this system with a relative error which is close to the error in the free term in the second equation. As is evident from the table, the uranium content is determined with an error which is practically the same as the error in its determination from the system with optimum threshold discrimination. However, the "differential" system considered has an essential advantage over systems with threshold discrimination, since only hard radiation is used in it. Consequently, the result of the measurement is not altered by the neglect of the changes in the coefficients of the equation due to the change in the composition of the rock, and depends less on the geometry of the measurement (in a surface observation and in blast holes with a nonuniform emitter content in the surrounding rock).

The ranges of the spectrum which were used in setting up the equations of this system were chosen so that the densities of the spectral distribution of secondary β-rays at the lower and upper boundaries of each of the ranges were as close as possible to each other; this causes a decrease in the error due to inaccurate positioning of the spectrometer window on the energy scale.

After considering what has been said, one can recommend the system of equations considered and its corre-

sponding system of two-channel measurement of the differential radiation intensity as being the most complete answer to the problem of the separate determination of uranium and thorium content in rock. Let us emphasize that the reduction in counter efficiency in a transition from measuring integral radiation intensity to differential is taken account of in the calculation given above for the accuracy of the measurements. Therefore the lower efficiency cannot serve as an argument against the practical application of the scheme considered.

For observations in blast holes, drilled holes, and for surface testing of rock with portable apparatus, a single scintillation counter using a scintillator weighing about 50 g (for which the calculations given above were used) gives enough accuracy to solve most practical problems. Depending on the type of problem being solved, the efficiency of the radiation measurement (using portable apparatus) should be increased by a factor of from 10 to 100; however, in this case the increase in efficiency must be attained by increasing the number of counters and the scintillator size, not by increasing the measured bandwidth of the spectrum.

For purposes of comparison, we list the accuracy obtained by other methods of making a separate determination of emitter content (with respect to γ-emission) which have been discussed in the literature. Thus, a value $m_U \approx 26\%$ is obtained for the apparatus of Gol'bek et al. [12, 13], using the most favorable discrimination level (at the same levels of observation) according to formula (3).

In this way, the accuracy of the separate determination using the scintillation γ-spectrometer with the given parameters is almost four times as large as the accuracy obtained with the Gol'bek apparatus.

According to theoretical results [3], measurements using copper and lead or tungsten MS-4 and VS-4 counters (without using the $\gamma\beta$-coincidence circuit) give a result with an error $m_U = 53\%$. In accordance with previous results [3], such measurements must be acknowledged as being unsuitable for the purpose of making a separate determination of emitter content.

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Received May 10, 1958

GAMMA RAYS IN AN ATOMIC EXPLOSION

O. I. Leipunskii

The question of γ-radiation in an atomic explosion is discussed. The physical factors which are essential for the appearance of γ-radiation are analyzed. The influence of a shock wave and atmospheric heating on the passage of γ-rays are introduced into the calculation, as is the presence of capture gamma-rays.

A calculation is made of the γ-ray dosage, which, because of the effect of the shock wave, increases at a more rapid rate than proportionally to the explosion energy. It is shown that in the course of time the dosage rate passes through a minimum and a maximum. A hard component is present in the γ-radiation and appears at great distances.

The analysis of the phenomena accompanying an atomic explosion leads to problems which are of interest from a physical point of view. In the present discussion the question of the γ-ray dosage in an atomic explosion is considered.*

Statement of the Problem

Given that the intensity of the monoenergetic radiation in a certain place is $J(t)$ Mev/sec·cm², we have for the dosage rate, $P(t) = 1.48 \cdot 10^{-5} \mu_{0E} J(t)$, , and for the dose after a time t

$$D = \int_0^t P(t)\,dt = 1{,}48 \cdot 10^{-5} \mu_{0E} \int_0^t J(t)\,dt, \qquad (1)$$

where $1{\cdot}48 \cdot 10^{-5}$ r/Mev is the conversion factor ‡ for changing the quantity of energy absorbed in Mev to roentgens; μ_{0E} cm^{-1} is the energy absorption coefficient of the γ-rays.

The system used in this problem of obtaining the dosage is as follows. At the point of the explosion, virtually a point source of γ-rays is originated, in the form of fission fragments (whose activity $G(t)$ Mev/sec and spectrum ϵ_0 are known). The gamma rays emitted in fission processes are not considered, since they are absorbed in the structure of the bomb. The gamma rays emitted by the fission fragments after the vaporization of the bomb exert their effect until the ascent with the incandescent air, i.e., for a period of several tens of seconds. The propagation of the γ-rays takes place in the practically infinite medium of air (the influence of the earth is not taken into account). Since a quantum can undergo only Compton collisions up until its absorption due to a photoeffect, the spatial intensity distribution of the γ-radiation $J(R, t)$ corresponds to that of a diffusion model of propagation of the quanta. The calculation of the multiple scattering of the quanta in the problem of the spatial intensity distribution or dose rate is done either by testing models [1]-[3], or by solving the integro-differential transport equation of the quanta, using electronic computers [4]-[6].

In the measurements of Leipunskii and Sakharov [1], the results are interpreted in the form of the formula

$$P(R) = \frac{1{,}48 \cdot 10^{-5} \mu_{0E} G a e^{-R/\lambda_{\text{eff}}}}{4\pi R^2} \text{ r/sec} \qquad (2)^{\S}$$

in which the calculation of the multiple scattering is carried out with the coefficients α and λ_{eff}, (which are functions of the distance R and the original energy of the quanta ϵ_0. In formula (2) λ_{eff} is the effective length of absorption of energy of the γ-rays under the conditions of multiple scattering (the fractional decrease in the dose is $1/e$ in the distance λ_{eff}; $\lambda_{\text{eff}} > \lambda_0$, where λ_0 is the mean free path of a quantum); a is a dimensionless coefficient. These coefficients depend only slightly on R, so that one can choose distance intervals several free paths long, in which α and λ_{eff} can be considered as constants, independent of R.

Values of α and λeff are listed in Table 1 for air for two distance intervals and initial quantum energies equal to 2 Mev ($\mu_{0E} = 3 \cdot 10^{-5}$ cm^{-1}) and 6 Mev.

Table 1.

Coefficients of Exponential Formula (2) for the Attenuation of γ-radiation in the Atmosphere

	Distance, interval, 400 m < R < 1000 m		Distance, interval, 1000 m < R < 2000 m	
	γ-ray energy, Mev		γ-ray energy, Mev	
	2	6	2	6
α	1,82	1,22	3,9	1.4
$\lambda_{\text{eff}}/\lambda_0$	1,31	1,25	1,13	1,15
λ_{eff}	$231 \frac{\rho_0^*}{\rho^{**}}$ m	$410 \frac{\rho_0}{\rho}$ m	$200 \frac{\rho_0}{\rho}$ m	$375 \frac{\rho_0}{\rho}$ m

* ρ_0 density of the air at standard temperature and pressure

** ρ density of the air

*The γ-ray dosage is the quantity of γ-ray energy absorbed in 1 cm³ of air. The size of the dose determines the severity of the biological consequences of the exposure. Since the energy absorbed is used ionizing atoms, it is proportional to the number of ions formed. Therefore the dose is usually measured in units of the quantity of electricity formed in the atmosphere upon irradiation. The dosage unit 1 ρ corresponds to 1 CGSE† quantity of electricity of each sign formed per cm³ of air at normal temperature and pressure. For a dose of 400 ρ, the mortality rate due to radiation sickness amounts to 50%.

† CGSE ≡ cgs electrostatic unit.

‡ One roentgen corresponds to an absorbed energy of

$$\frac{1 \text{ CGSE} \cdot 32{,}5 \cdot 10^{-6} \text{ Mev}}{P \cdot 4{,}8 \cdot 10^{-10} \text{ CGSE} p} = \frac{1}{1{,}48 \cdot 10^{-5}} \text{ Mev/r},$$

where $32{,}5 \cdot 10^{-6}$ Mev is the average energy spent in the formation of an ion pair.

§ Expression (2) is formally equivalent to a theoretical expression due to Fano [4]

$$P(R) = \frac{1}{4\pi R^2} 1{,}48 \cdot 10^{-5} \mu_{0E} G B(R/\lambda_0) e^{-R/\lambda_0},$$

but in place of two of the coefficients in the Fano formula (the accumulation factor B, which depends critically on the distance, and the mean free path λ_0, which is independent of distance) two other coefficients which vary little with distance are introduced: α and the effective mean free path λ_{eff}. In many cases it is preferable to use formula (2).

Expression (2), with these values of γ_{eff} and α is correct to within 10%. It would seem that one could find the gamma-ray dosage by substituting expression (2) in (1), using the known $G(t)$ and ε_0.

However, such a method is inadequate. In a discussion of the present problem, Ya. B. Zel'dovich and the author noted that the absorption of γ-rays along the path from the center of the explosion to the point of observation must be sensitive to the redistribution of mass of the atmosphere occasioned by the passage of the shock wave. A calculation of the passage of the γ-rays must take into account the actual density distribution of the air which is set up by the shock wave during the time of emission of the γ-rays.

In the present work, an estimate is made of the influence of the shock wave on the formation of the dose. The method of calculation of the influence of the shock wave was proposed by Ya. B. Zel'dovich.

Influence of the Shock Wave

Expression (2) is basic in the dose rate calculation. The value 2 Mev, corresponding to two observed energies (1.6 [7] and 2.2 Mev [8]) is taken as an average energy of the γ-emission of the fragments. In calculating the density distribution of the air, an "optical thickness" for γ-ray absorption is introduced. It is known that if the absorption of the radiation follows an exponential law, the intensity of the radiation which has covered a distance R in an inhomogeneous absorbing medium with an absorption length $\lambda(R)$, is determined by the so-called optical thickness

$$x = \int_0^R \frac{dR}{\lambda}. \tag{3}$$

The attenuation of the radiation corresponding to a distance R is equal to e^{-x}.

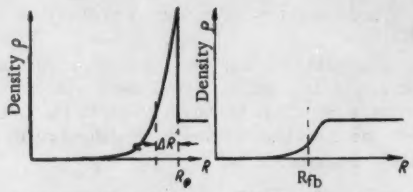
Because of the presence of multiple scattering, the γ-ray absorption behaves somewhat differently from an exponential. However, it can be represented as an exponential in determining the distance interval. This enables one to extend the concept of "optical thickness" to cases of γ-ray absorption in an inhomogeneous medium, and to consider that the attenuation* of the intensity due to absorption amounts to†

$$\exp(-x) = \exp\left(-\int_0^R \frac{dR}{\lambda_{eff}}\right) = \exp\left(-\frac{1}{\lambda_{eff}(\rho_0)\rho_0}\int_0^R \rho dR\right). \tag{4}$$

The optical thickness along a path R in the undisturbed atmosphere is equal to R/λ_{eff}. Let the radius of a wave front of a strong‡ spherical shock wave be R_Φ. Practically all of the air in the strong shock wave is concentrated in the outermost thickness ΔR at the surface of the shock wave ($\Delta R \ll R_\Phi$, Fig. 1, *a*). The optical thickness equals

$$\frac{1}{\lambda_{eff}(\rho_0)\rho_0}\int_0^R \rho\, dR = \frac{1}{\lambda_{eff}}\frac{\rho\Delta R}{\rho_0} \tag{5}$$

(the approximation is made that $\Delta R\rho = \text{const}$ inside the interval).



a) *b)*

Fig. 1. Density distribution of the air in a shock wave.

We shall find the value of the product $\rho\Delta R/\rho_0$ from the condition that the mass of air in the edge of a surface of radius R_Φ is equal to the mass of air which was in a sphere of R_Φ until the shock wave was formed:

$$\rho 4\pi R_\Phi^2 \Delta R = \rho_0 \frac{4}{3}\pi R_\Phi^3,$$

$$\frac{\rho}{\rho_0}\Delta R = \frac{R_\Phi}{3}.$$

According to formula (5),

$$x = \frac{1}{3}\frac{R_\Phi}{\lambda_{eff}}.$$

In the absence of a strong shock wave, the optical thickness equals R/λ_{eff}. Thus the redistribution of the atmospheric mass in a strong shock wave results in a decrease of two thirds of R_Φ/λ_{eff} in the optical thickness along a path R. The γ-ray absorption inside the shock wave proceeds as if it contained a cavity of radius L:

$$L = R - \frac{\int \rho\, dR}{\rho_0}. \tag{6}$$

In the case of a strong shock wave

$$L = \frac{2}{3}R_\Phi. \tag{6a}$$

The absorption of the γ-rays occurs outside of the cavity, in a thickness ΔR of the air at the wave front:

$$\Delta R = \frac{R_\Phi}{3}\frac{\rho_0}{\rho}. \tag{6b}$$

Thus the optical thickness equals $(R-L)/\lambda_{eff}$ at a distance R from the center of the explosion ($R > L$). The radiation intensity will be proportional to

$$\exp[-(R-L)/\lambda_{eff}],$$

since in the absence of a shock wave the intensity would be proportional to

$$\exp(-R/\lambda_{eff}).$$

The formation of a strong shock wave along the path of propagation of the γ-rays increases the radiation intensity by a factor (L/λ_{eff}). At an instant at which R_Φ is equal to 700 m (and for a sufficiently large bomb

*The geometric attenuation of intensity is expressed by the $1/4\pi R^2$.

†In a two-dimensional problem the concept of "optical thickness" in connection with the γ-ray absorption calculation is exact, while it is approximate in the spherical case. In experiments made by Leipunskii and Sakharov [1], it was shown that one could use this approximation for $R > \lambda_{eff}$ and near the boundary of a pronounced inhomogeneity. For radiation with a purely exponential absorption, the application of the optical thickness is exact in the spherical case also.

‡A strong shock wave is characterized by a pressure at its front which is much higher than atmospheric.

energy, so that the shock wave will still be strong), the radiation intensity will be increased by a factor of seven:

$$e^{L/\lambda_{\text{eff}}} = e^{\frac{2}{3}\cdot\frac{700}{230}} = 7.$$

The effect which the shock wave has on the γ-ray absorption also extends to the time at which the shock wave has gone a great distance from the center and has become an acoustic wave. At this stage, the pressure in the region of the explosion becomes equal to the atmospheric pressure p_N. However, the density of the air does not become equal to the density before the explosion. Since the passage of the shock wave is accompanied by the subsequent heating of the air ([9], p. 207, Fig. 5) [10], the air becomes incandescent in the region where the shock wave was strong, due to the large residual heating. The fire ball, which is formed after the explosion, appears immediately after the strong shock wave which had passed earlier. The pressure in the fire ball is equal to atmospheric, but the density is low because of the high temperature (Fig. 1,b).

Thus, the air density is small in the strong shock wave region both during the passage of the shock wave and afterward, i.e., during the whole period of emission of the γ-rays. The radius of the fire ball R_{fb} is fixed by the radius of the outermost wave front of the strong shock wave which had passed previously.

Measurements of the propagation γ-rays in water carried out by Leipunskii and Sakharov confirm that the presence of the cavity of radius L surrounding the source causes the intensity at a distance $R\,(R > L)$ to increase by a factor of $\exp(L/\lambda_{\text{eff}})$. Thus, the radiation intensity is determined by formula (2) which has $R-L$ in the exponent instead of R. The coefficient α is practically constant in this instance. Instead of formula (2), we get

$$P(R) = \frac{1{,}48\cdot 10^{-3}\mu_{0E}G\alpha e^{L/\lambda_{\text{eff}}}e^{-R/\lambda_{\text{eff}}}}{4\pi R^2}. \qquad (7)$$

Estimate of the Dosage

According to expressions (1) and (7), the dose at a distance R from the center of the explosion is found according to the formula

$$D = \int_0^{t_{\text{eff}}} P\,dt = \frac{1{,}48\cdot 10^{-3}\mu_{0E}\alpha e^{-R/\lambda_{\text{eff}}}N}{4\pi R^2\cdot 10^4} \times \int_0^{t_{\text{eff}}} u(t)\,e^{\frac{L(E,t)}{\lambda_{\text{eff}}}}\,dt, \qquad (8)$$

where $G(t) = Nu(t)$ Mev/sec; $N = 1.45\cdot 10^{23}$ is the number of fissions; E_{TT} is the energy of the explosion expressed in thousands of tons of TNT (1 thousand tons corresponds to 10^{12} cal. $= 4{,}17\cdot 10^{19}$ erg); $u(t)$ Mev/sec · bel is the intensity of the energy emitted by the fission fragments per act of fission; t_{eff} is the effective time during which one must consider the effect of the γ-rays (because of the radioactive decay of the fragments and the ascent of the cloud of fragments, the γ-radiation intensity drops rapidly) and which is taken equal to 10 sec; 10^4 is the coefficient for R^2 in square meters. The integral in expression (8) can be put in the form

$$\int_0^{t_{\text{eff}}} u(t)\,e^{\frac{L(E,t)}{\lambda_{\text{eff}}}}\,dt = e^{\frac{\overline{L}(E)}{\lambda_{\text{eff}}}} \times \int_0^{t_{\text{eff}}} u(t)\,dt = e^{\frac{L(E)}{\lambda_{\text{eff}}}}\overline{W}, \qquad (9)$$

where

$$\overline{W} = \int_0^{10\text{ sec}} u(t)\,dt = 2{,}05 \text{ Mev/fis} \quad [11].$$

By substituting the numerical values of the coefficients in expressions (8) and (9), we obtain

$$D = \frac{2{,}0\cdot 10^3 E_{TT}e^{\overline{L}/\lambda_{\text{eff}}}e^{-R/\lambda_{\text{eff}}}}{R^2}\,p, \qquad (10)$$

where R is in meters.

The estimate $\overline{L}$ is made according to the following assumptions. It is assumed that $\overline{L}$ is close to the final radius of the fire ball – the region of incandescent air formed after the explosion: $\overline{L} \approx R_{fb}$. The approximation made becomes more accurate the smaller the time of formation of the final dimensions of the cavity in comparison with time of γ-ray emission (inasmuch as the expression with $\overline{L} \approx R_{fb}$ is constant and taken outside of the integral over time), i.e., it is more accurate, the less the energy of the explosion. The approximation made is known to be false for explosions of very large energy, since a large part of the γ-radiation is emitted until the fire ball acquires its final size, i.e., for large explosion energies ($\overline{L} < R_{fb}$). The final radius of the fire ball can be estimated in accordance with the amount of energy contained in it. It follows from a theory proposed by Sedov ([9], p. 207, Fig. 56), that the energy of the incandescent air comprises about 50% of the energy of the explosion.

Then

$$\frac{4}{3}\pi R_{fb.\,m}^3\,\epsilon = 0{,}5E,$$

$$\overline{L} \approx R_{fb.\,m} = \sqrt[3]{\frac{0{,}5E}{\frac{4}{3}\pi\epsilon}}, \qquad (11)$$

where ϵ is the energy in units of the volume of the fire ball. Since the energy of 1 g of air is

$$\frac{5R(T-T_N)}{28{,}8} = 0{,}35\,(T-T_N) \text{ cal/g},$$

then we have (neglecting T_N in comparison with T)

$$\epsilon = \rho\cdot 0{,}35\,T = \frac{\rho_0 T_0}{p_0}\cdot\frac{p_N}{T_N}0{,}35\,T = 1{,}23\cdot 10^{-1}\,p_N \text{ cal/cm}^3,$$

where p_N is the pressure of the air before and after the explosion (atm); T_N is the temperature of the air before the explosion; T is the temperature of the air after the explosion; ρ_0, T, p_0, are the density, temperature and pressure of the air under standard conditions.

Substituting ϵ in formula (11), we obtain

$$\overline{L} = 100\sqrt[3]{\frac{E_{TT}}{p_N}} \text{ m}. \qquad (12)$$

Substituting (12) in (10), we obtain

$$D = \frac{2{,}0\cdot 10^3 E_{TT}e^{100\sqrt[3]{\frac{E_{TT}}{p_N}}e^{-R/\lambda_{\text{eff}}}}}{R^2}. \qquad (13)$$

* The increase in heat capacity with temperature and dissociation is neglected in this estimate.

For the case in which $p_N = 1$ atm, $T_N = 25$ deg C,

$$\lambda_{eff} = 231\frac{298}{273} = 252 \text{ m},$$

$$D = \frac{2{,}0\cdot 10^9 E_{TT} e^{0{,}4\sqrt[3]{E_{TT}}} e^{-R/252}}{R^2} = \frac{20\cdot 10^9 E_{TT} 10^{0{,}174\sqrt[3]{E_{TT}}} e^{-R/252}}{R^2}. \quad (13a)$$

The factor

$$\exp(\bar{L}/\lambda_{eff}) \cong \exp\left(\frac{100\sqrt[3]{E_{TT}/p_N}}{\lambda_{eff}}\right),$$

which specifically appears in expressions (10), (13) and (13a), we shall refer to as the cavity factor. This factor does not appear in ordinary problems involving the propagation of γ-rays. It appears in the explosion problem because the source of γ-rays itself (the explosion) causes the absorbing power of the medium to change during the emission. It is important to note that according to (10), (13), the dose at a given distance from the center does not increase linearly with the energy, but rather at a more rapid rate (proportional to $E_{TT} e^{0{,}4\sqrt[3]{E_{TT}}}$).

If the effect of the shock wave is disregarded, the dosage must be proportional to the number of fragments, i.e., the number of acts of fission, or the explosion energy E_{TT}. Also, the amount of the dose becomes smaller by the cavity factor, i.e., by a factor of $e^{0{,}4\sqrt[3]{E_{TT}}}$ (for the case of a nominal bomb, in which E_{TT}=20,000 tons, the factor is 3). For explosions of higher energy, the cavity factor will become still more important, and its neglect will lead to order-of-magnitude errors.

In an article by Lewis [13], values of dosage are listed for various distances from the epicenter of the explosion at Hiroshima, i.e., for the case of a nominal bomb(E_{TT} = 20,000 tons). It is therefore possible to compare the results of theoretical estimates with experimental data.

For the case of a nominal bomb, formula (13) takes the form

$$D = \frac{1{,}2\cdot 10^{11} e^{-R/252}}{R^2} \text{ r}. \quad (14)$$

Values of γ-ray dosages in the explosion of the Hiroshima bomb are listed in Table 2 in agreement with Figure 1 of the Lewis article [13].* The data of Table 2 are used in plotting Fig. 2. It is clear from the figure that for the distances in the range 900-1600 m for which the dosages were high (greater than 50 r), the empirical formula

$$D = \frac{6{,}6\cdot 10^{10} e^{-R/252}}{R^2} \text{ r}. \quad (15)$$

is correct.

For purposes of comparison with the theoretical estimate of formula (14), we shall reduce formula (15) to a form in which the value of λ_{eff} is the same as in (14), i.e., 252 m.

Formula (15a), which corresponds to the dashed line in Fig. 2, can be used to replace formula (15), with an error which is zero at a distance of 1100 m and which does not exceed ± 20% at the ends of the distance interval referred to.

$$D = \frac{1{,}1\cdot 10^{11} e^{-R/252}}{R^2} \text{ r}. \quad (15a)$$

Table 2.

γ- Ray Dosages from the Explosion of the Hiroshima Bomb

Distance from epicenter, m	Distance from center, m	Dose, r
740	950	2500
1000	1170	800
1250	1390	250
1500	1620	90
2000	2090	13
2500	2580	2,4

This empirical formula is practically the same as the theoretical estimate (14). It should be pointed out that one cannot expect good agreement with experiment for large explosion energies, since in this case it is impossible to calculate $\bar{L}$ by formulas (11), (12) which are correct for the final radius of the fire ball. As was shown above, $\bar{L} < 100\sqrt[3]{\frac{E_{TT}}{p_N}}$ for high energy. $\bar{L}$ must also in this case increase with energy more slowly than $\sqrt[3]{E}$ (namely, as $\sqrt[5]{E}$), since the radius of the front of the strong shock wave at a given instant varies as $\sim \sqrt[5]{E}$ [9], p. 197, formula 7.4).

As was shown above, using as a value for $\bar{L}$ the limiting radius of the strong shock wave (proportional to $\sim\sqrt[3]{E}$) is a working approximation which is applicable in a case in which the time taken by the strong shock wave to attain its maximum radius is considerably less than the time during which the γ-rays are emitted.

Intensity of the γ-Radiation (Dosage Rate)

Were it not for the shock wave, the dosage rate would decrease monotonically in time, proportionally to the activity of the fission fragments (monotonically decreasing because of the radioactive decay). However, because of the formation and growth of the cavity, the γ-ray absorption decreases in time, and when the shock wave passes through the point at which measurements are taken, this point appears to be in the cavity and the absorption of γ-rays will practically be absent (if the shock wave is strong in this place).

Thus the dosage rate must decrease due to the decay of the fragments and the ascent of the cloud of fragments; and must increase due to the growth of the cavity. As a result of the superposition of these two processes, the measured dose rate must start to increase and pass through a maximum.

P. A. Yampol'skii has expressed the thought that besides the radiation from the fission fragments, there might appear initially more γ-radiation from the radiative neutron capture by atmospheric nitrogen in the reaction $N^{14}(n,\gamma)N^{15} + 10.88$ Mev [14]. The average energy of this γ-radiation is ~6 Mev. Radiative capture

* The distances from the center of the explosion were calculated on the assumption that the explosion took place at an altitude of 1000 ft (/12/, sect. 5.45 and 11.5). The total dosages from neutrons and γ-rays were listed in the article by Lewis. Inasmuch as "neutrons which are produced in an explosion do not create additional danger" (/12/, sect. 7.64), the figures in the Lewis article are, in fact, γ-ray dosages.

accounts for $^1/_{17}$ of the evolved neutrons (the remaining $^{16}/_{17}$ disappears in the $N^{14}(n,p)C^{14}$ reaction, which is not accompanied by γ-emission). Since 1.5 excess neutrons are liberated per each act of fission, the amount of energy of the capture radiation incident per act of fission equals $\frac{10{,}88 \cdot 1{,}5}{17}$ 1 Mev/fis. The number of neutrons in the atmosphere decreases exponentially with

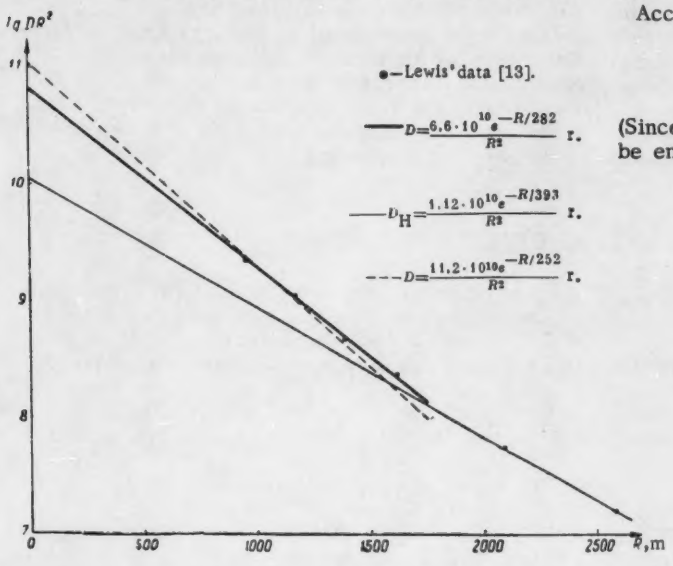


Fig. 2. γ-ray dosage in the explosion of the Hiroshima.

a period equal to the life time of neutrons in air, i.e., 0.06 sec. Therefore the initial activity of the capture radiation must be

$$\frac{1 \text{ Mev/fis}}{0{,}06 \text{ sec}} = 16 \text{ Mev/sec} \cdot \text{fis},$$

which is 20 times as large as the initial activity of the fragments (0.8 Mev/sec·fis [11]).

By the time several periods have passed, i.e., within 0.2-0.3 sec after the explosion, the capture radiation can be neglected. Figure 3 illustrates explicit qualitative considerations on the form of the time dependence of the dosage rate.

The Hard Component in the γ-Ray Dosage

The decrease in γ-ray dosages with increase in distance (in reference to the Hiroshima explosion and the article by Lewis which was cited above) is shown in Fig. 2. The presence of a hard component in the γ-ray spectrum is evident in the figure: the slope of the curve $\log DR^2 = f(R)$ decreases with increasing distance. λ_{eff} = 282 m in the range of distances from 600 to 1600 m, and in the distance interval from 2000 to 2600 m, λ_{eff} = 383 m.

A value of the γ-radiation energy of about 6 Mev corresponds to the latter value of λ_{eff}. We do not know whether there is radiation of energy greater than 2.2 Mev included in the γ-radiation of the fission fragments. Until this question is answered we can connect the presence of a hard component only with capture radiation which has a nearby value of mean energy (see the preceding section).

The dosage of the hard component, in accordance with Fig. 2, can be expressed by the empirical formula

$$D_H = \frac{1{,}1 \cdot 10^{10} e^{-R/393}}{R^2} \text{ r}. \qquad (16)$$

We will derive an estimate of the hard component dosage by assuming that it is formed only by the capture radiation.

According to formula (7)

$$D_H = \frac{1{,}48 \cdot 10^{-5} N \varepsilon \alpha \mu_{0\gamma} e^{-R/\lambda_{eff}}}{4\pi R^2 \cdot 10^4} \text{ r}. \qquad (17)$$

(Since the principal part of the capture radiation must be emitted during a time of 0.1 sec, when the radius

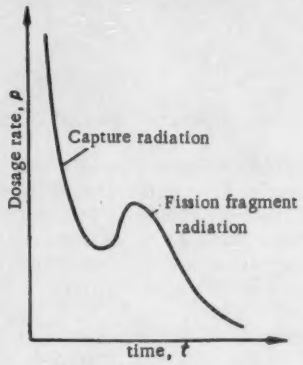


Fig. 3. Time variation of the γ-ray dosage rate.

of the shock wave is still small, the cavity factor is not taken into account.) Substituting in formula (17)

$$N = 1{,}45 \cdot 10^{23} E_{TT} \text{ (for } \varepsilon = 1 \text{ Mev/fis)};$$
$$\left.\begin{array}{l} \alpha = 1{,}4 \\ \lambda_{eff} = 410 \text{ m} \end{array}\right\} \; 1000 \text{ m} < R < 2000 \text{ m (Table 1)};$$
$$\mu_{0\gamma} = 1{,}9 \cdot 10^{-5} \text{ cm}^{-1} \text{ (for } \varepsilon_0 = 6 \text{ Mev)};$$
$$E_{TT} = 20{,}000 \text{ ton, we obtain}$$

$$D_H = \frac{9{,}1 \cdot 10^{9} e^{-R/410}}{R^2} \text{ r}. \qquad (17a)$$

The calculated estimate is found in satisfactory agreement with the empirical formula (16). This bears out the proposal that the hard component comes from neutron capture by atmospheric nitrogen.

Summary

1. The γ-ray dosage in an atomic explosion depends not only on the quantity of γ-rays emitted, but also on the shock wave which accompanies the explosion and redistributes the density of the air so that the γ-ray absorption is decreased and the dosage increased.

The shock wave forms a cavity in the atmosphere in which absorption of γ-rays is absent. The cavity dimensions increase with the energy of the explosion. Therefore the increase with energy of the γ-ray dosage at a fixed distance from the center of the explosion is faster than it would be for a direct proportion between dosage and energy. The increase in dosage due to the action of the shock wave is expressed by a number, the cavity factor, with which one can arrive at orders of magnitude.

A quantitive estimate of the dosage is expressed by the formula

$$D = 2{,}0 \cdot 10^9 E_{\tau\tau} e^{0{,}4\sqrt{E_{\tau\tau}}} \frac{e^{-R/252}}{R^2} \text{ r.}$$

An estimate with this formula yields values which are close to the dosages which occurred at Hiroshima.

2. In addition to the fission fragment radiation, γ-radiation from neutron capture by nitrogen appears. As a result of the superposition of the natural drop in radiation intensity in time and the increase in transparency of the atmosphere for γ-rays because of the growth of the cavity, the γ-ray dosage rate passes through a minimum and a maximum.

3. There is a hard component (with energy about 6 Mev) in the composition of the γ-rays which make up the dosage in an explosion. Starting from the identification of the hard component with the capture radiation, we obtained an estimate of the dosage of the hard component

$$D_{\text{H}} = 4{,}6 \cdot 10^8 E_{\tau\tau} \frac{e^{-R/410}}{R^2} \text{ r.}$$

An estimate by this formula yields values close to the dosages which occurred at Hiroshima.

The author expresses his deep gratitude to Ya. B. Zel'dovich for his interest and many suggestions. The author also expresses his deep gratitude to P. A. Yampol'skii and V. N. Sakharov for their fruitful participation in a discussion of the problem and an analysis of the physical phenomena.

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MEASUREMENT OF TISSUE DOSES OF HARD BREMSSTRAHLUNG

B. M. Isaev and M. I. Shal'nov

Using a synchroton generating gamma-quanta at an energy of up to 250 Mev, a study was made in a paraffin phantom of the distribution of penetrating doses at energy values of 80, 180, and 250 Mev.

A close approximation to the position of the ionization maximum is given by the expression $d/\sqrt{E} = 1{,}14(1 - e^{-0{,}1E})$, for $E > 4$ Mev, and for a focusing distance of 100 cm, where E is the maximum energy of the bremsstrahlung (Mev), d the depth of the maximum (g/cm^2).

In this article methods are discussed of measuring the average tissue dose of hard bremsstrahlung in rads using thimble ionization chambers and calorimeters.

Measurement of the tissue dose of hard penetrating radiation was carried out with the use of a phantom, which was made up out of separate blocks measuring 30 × 30 cm and having thicknesses of 1, 2, 3, and 5 cm and forming a total thickness along the path of the beam of 41 cm. X-ray films measuring 24 × 30 cm were placed in paper cassettes between the paraffin layers. The front edge of the phantom was at a distance of 260 cm from the target.

The depth of darkening of the x-ray film irradiated in the phantom was measured on a densitometer over the whole area of the spot, and compared with the corresponding depth in control films irradiated with γ-rays of Co^{60}.

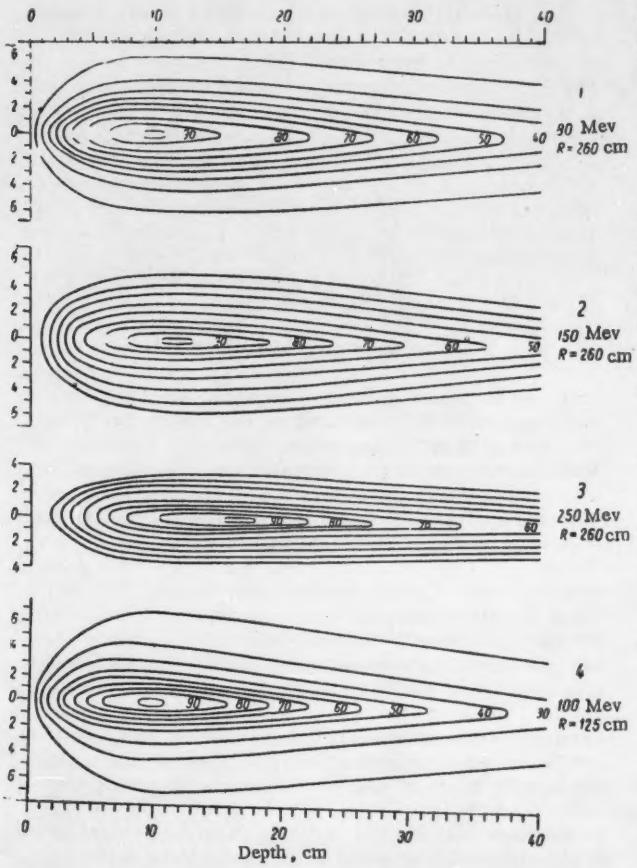

Fig. 1. Distribution of deep doses in a paraffin phantom with maximal energies (80, 150, and 250 Mev) of bremsstrahlung. R is distance from target to front face of phantom.

The three-dimensional distribution of penetrating doses in a paraffin phantom at energies of 80, 150, and 250 Mev, as determined by this method, is illustrated in curves 1, 2, and 3 of Fig. 1. For comparison we give the relative distribution of penetrating doses (isodose curves) obtained by Charlton and Bread [1] in a pressboard phantom at a maximum quantal energy of 100 Mev and a focusing distance of 125 cm (curve 4).

The distribution of the penetrating doses along the axis of the bundle is shown in the curves of Fig. 2, where for comparison we also give the curves of penetrating doses at 20, 30, and 50 Mev, obtained in [1].

As would be expected, the penetrating dose attains a maximum at a certain distance from the surface of the object characteristic of the energy of the incident -quanta, after which the primary beam gradually decreases in intensity as it is absorbed. The depth of the layer of substance corresponding to the maximum absorption of energy represents an approximate value for the free flight of secondary electrons in the substance.

As is well known, the presence of this maximum plays an essential part in medical radiology, because at high energies the absorption of energy is small at the skin, and gradually increases at greater depths in the tissue. The position of the maximum in the case of irradiating with a definite energy (hardness) depends on the focusing distance and the density of the substance used. Mayneord and Martin [2] have compared the results of different authors by referring their results to a substance of unit density and to a common focusing distance of 100 cm. The correction for the displacement of the maximum at a focusing distance of 260 cm, instead of 100 cm, can be shown by extrapolation not to exceed 25-30%.

If our results are referred to a focusing distance of 100 cm, involving a correction of 25%, then they agree quite well with the curves constructed by Mayneord and Martin (Fig. 3). The expression representing our curve has the form:

$$d/\sqrt{E} = 1{,}14(1 - e^{-0{,}1E}),$$

where E is the maximum energy of the inhibitory radiation, and d the depth of the maximum.

This expression is almost correct for all energies 4 Mev. For energies > 50 Mev

$$d/\sqrt{E} = \text{const} = 1{,}14.$$

The curve of Fig. 3 may be used to determine the position of the maximum in the beam. For a precise determination of the energy absorbed by any substance (in ergs/g), the Bragg-Gray conditions for secondary particles must be strictly fulfilled in measurements by the ionization method.

In practice, this condition can be met by an assembly of thick layers of appropriate material. This method

can be used, for example, in measurements on a beam of high energy neutrons [3].

For measurements on a beam of hard bremsstrahlung, Mayneord and Martin [2] used coatings of magnesium, graphite, copper, lead, and other materials which were placed over thin-walled chambers. The walls of the chambers themselves were made of graphite, copper, or lead, or of paraffin impregnated with graphite.

The curves obtained on increasing the thickness of the chamber walls to some extent reproduce the curves of the depth distribution of the energy absorbed in the phantom (Fig. 2).

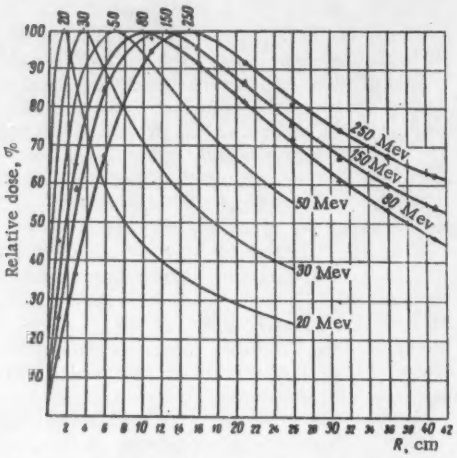

Fig. 2. Distribution of deep doses in a paraffin phantom along the axis of the beam for different energies of bremsstrahlung.

Even for energies in the 20-30 Mev range, the magnitude of the dose in roentgens in the beam is measured with an accuracy which is sufficient only for the comparison of the power of different installations, but is not sufficient to describe their biological effects. Evidently, the error in such measurements will increase in energy, so that the method can hardly be applied for energies in excess of 30 Mev.

Recently, most authorities have inclined to the view that dosimetry should take the course of making absolute energy measurements in ergs/g or in rads (1 rad = 100 ergs/g).

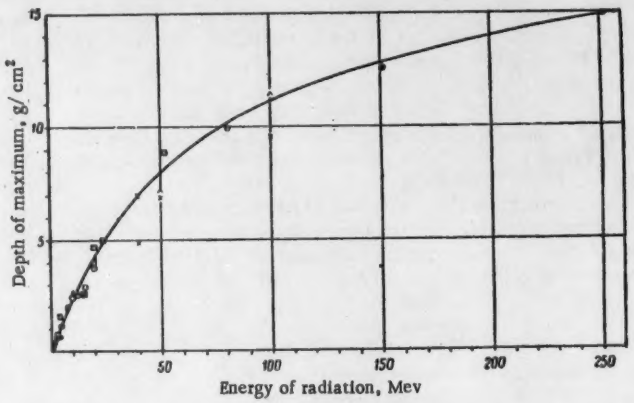

Fig. 3. Position of the maximum absorption of energy along the axis of a beam for different energy γ-quanta of bremsstrahlung: ∧—results obtained by Failla, Bristrup, Kleber, Charlton, Bread; ⊛—results obtained by Kerst; ×—results of Charlton, Bread; •—results of Trump; □—results of Mayneord and Martin; ⊕—results of Isaev, Shal'nov.

It is particularly important to note the following: Figures obtained by measurements with a graphite chamber on increasing the thickness of the walls, agree well with measurements on the phantom; the maxima of the corresponding curves are shown in the curve of Fig. 3. From this it can be deduced that graphite is a suitable material for the chamber walls.

It is necessary to decide in what units doses of high energy electromagnetic radiation are to be measured.

All the conditions of the definition of the roentgen are fulfilled in x-ray and γ-ray dosimetry up to energies of approximately 2 Mev. At high energies, it is not possible to satisfy all these conditions.

In this case, doses measured in air space at depths greater than the equilibrium thickness conform completely to the conditions of the definition of a roentgen. In order to establish conditions which satisfy the definition at high energies, it becomes necessary to increase the thickness of the walls of the chamber to a size equal to the range of an electron having maximal energy. But in this case, the unknown extent of the scattered secondary radiation is introduced into the calculation. Also, considerable difficulty is encountered in estimating the contribution of the secondary γ-irradiation resulting from the retardation of high speed electrons in the substance of the walls during the annihilation of positrons. These components, resulting from the interaction of penetrating γ-quanta with the substance are not included in the terms of the definition of the roentgen.

In the region of high energy radiations, for which there is as yet no one single method of measurement, the absorption of energy in absolute units must be determined by special means.

The recording of the depth distribution of ionization in a phantom forms a necessary part of such measurements. Here, measurements are made most economically by a photographic method, or, alternatively, an ionization chamber may be used. In the first case it is convenient to use a paraffin phantom (as was done in our experiments); in the second, a water phantom may be used, and mechanical means (a topograph) may be employed to move a probe carrying an ionization chamber over the whole of the irradiated volume (see Figs. 1 and 2). The degree of ionization at the different points is most conveniently expressed as a fraction of the maximal ionization which is taken as unity. Further, the relative measurements taken over the whole of the volume of the phantom are integrated. This integral represents a measure of the absorption of energy in the given region, and is expressed in arbitrary units. It must be noticed that the curves, showing the depth distribution of the ionization in the phantom, allow an estimate of the fraction of the energy which is not absorbed in the phantom to be made from the extrapolar part ("tail") of these curves. The ratio of the integral, calculated from the "tail" of the curves, to the complete integral (including the "tail" of the curves), gives the relative value of the energy not absorbed in the phantom ("straight through" correction). The calcu-

lated value of the integral must be converted into absolute units, and for this purpose either the ionization or the calorimetric method may be used.

Ionization Method

If we have a thin-walled thimble ionization chamber with air-equivalent walls, which satisfies the definition of the roentgen for normal energies and which is calibrated in roentgens for these energies, then the indication of such a chamber in the corresponding phantom will represent a measure of the most important quantity —that of the radiant energy absorbed per 1 g of the substance of the phantom.

Indeed, with such a form of measuring device, the observed ionization is an indication of the corpuscular radiation passing through the substance at the point where the chamber is placed (the thickness of the wall may be neglected if it introduces no change in the distribution of the ionization within the substance investigated).

It is known that the ionization in a small chamber represents a measure of the energy lost by the secondary particles in the element of volume surrounding this cavity. For our purposes it is a matter of indifference from which direction the particles arrive, and how they are distributed. The problem is only one of realizing the transition from ionization in a small air cavity to ionization in the surrounding substance.

Let us suppose that the ionization in the small cavity is measured in esu/cm^3. Under normal conditions of pressure, temperature and the relations of the gases in the air of the chamber, the ionization current in 1 esu/cm^3 corresponds to the formation of $1.62 \cdot 10^{12}$ pairs of ions per gram of air in the cavity, or to an absorption of energy of 84 erg/g of air. At the same time, the absorption of energy ΔE per gram of the substance surrounding the cavity at an ionization current in the room of 1 CGSE/cm^3*, is, from the fundamental Bragg-Gray principle equal to

$$\Delta E = 1{,}62 \cdot 10^{12} \rho\varepsilon \text{ ev/g}$$

or

$$\Delta E = 2{,}58\ \rho\varepsilon \text{ erg/g},$$

where ρ is the relative retarding power of the substance (as compared with air), and ϵ is the work involved in the formation of one pair of ions in air.

The product ρ , is evidently not constant for different electron energies. The quantity ϵ , taken as equal to 32.5 ev for the normal electrons of moderate energy, is even less precisely known for the high energy electrons.

The relative retarding power for small energies ($E_T < 5\ mc^2$) and for substances with small atomic numbers, such as graphite, is greater than unity, but for very high energies it increases in such a way that for substances with an atomic number less than that of air, it becomes equal to unity.

To standarize the relative values of penetrating doses into absolute units of absorbed energy, the measurement of the latter is best made in the region of the maximum.

If all measurements in the phantom are made using an ionization chamber, then it will be advantageous to calibrate it first in roentgens, using normal x-rays, and then the magnitude of the absorbed energy which it is required to find will be given by the integrated value of the curve of the depth distribution.

In using photographic film for relative measurements in the phantom, it is convenient to carry out measurements of the absorbed energy using an ionization chamber surrounded by an equivalent layer of the substance of which the phantom is made (to correspond to the point of the maximum); a special instrument—a monitor—is used to control the variations in the intensity of the radiation during the exposure of the films and the chamber. (Standardization is carried out taking into account the corrections indicated by the monitor.) In this case, the absorbed energy is found by multiplying the absolute magnitude of the energy absorbed at the maximum by the ionization integrated over the whole volume, as found in relative units from photographic measurements.

In both cases, the complete absorption of radiant energy is found in ergs. By dividing this energy by the mass of the substance for which the absorption of energy is calculated, we find the average tissue dose in erg/g. The ratio of this dose to the maximum dose will not be the same for different energies of radiation (Fig. 4). The curves in this drawing show that for large objects, 20-26 cm in thickness, the average tissue dose, starting from energies of about 40-60 Mev, becomes constant and equal to approximately 80% of the maximum dose. For small objects there is a noticeable "energy dependence." These curves can be used in practice by measuring the absorption of energy in absolute units using an ionization chamber having a wall thickness appropriate to the particular energy.

Thus, the average tissue dose is determined from the curves of Fig. 4 as the product of the maximal absorption of energy and the corresponding percentage. The maximal value of the dose is important in connection with localized irradiation of separate organs lying at the corresponding depth in a large object.

In measuring the maximal value of the dose, it is not absolutely necessary to surround the thimble chamber with a layer of material equivalent to the tissue; this can be done by using magnesium, copper, or lead in a layer whose thickness (in g/cm^2) is selected from the curve of Fig. 3. To the values obtained, a correction must be added which will depend on the atomic number of the materiai used. Since the relative effectiveness of ionization chambers increases with increase in the atomic number of the walls, and since the values recorded for the ionization in magnesium, copper, or lead

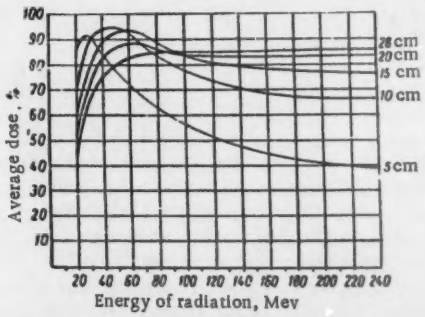

Fig. 4. Value of the average tissue dose relative to the maximal dose for different energies of bremsstrahlung (thickness of objects 5, 10, 15, 20, and 26 cm).

* CGSE ≡ cgs electrostatic unit.

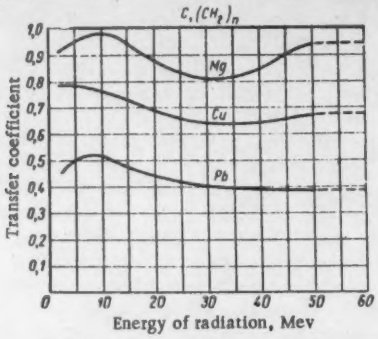

Fig. 5. Coefficients for transforming from ionization in magnesium, copper, and lead chambers to ionization in graphite, paraffin, and tissue.

will be higher than in the tissue-equivalent material, all the indications must be referred to the material. For this purpose, use may be made of the transfer coefficients of Fig. 5, which have been constructed from the experimentally determined points taken from [2] up to energies of 50 Mev.

For energies above 50 Mev, the values of the coefficients change by only a negligible amount, so that in practice they may be considered to be the same as at energies of 50 Mev.

Calorimetric Method

This method of measurement consists in determining, by means of a sensitive calorimeter, the flow of radiant energy (in ergs) falling on a completely illuminated surface of a large water phantom, and subsequently determining the relative distribution of the energy absorbed in this phantom by means of a probe fitted with a thimble ionization chamber. From the results of the relative ionization measurements in the phantom, the integral of the ionization is determined.

In calculations based on measurements with the calorimeter, a correction is made for the unabsorbed energy; this allows a precise value to be obtained for the whole incident energy flux of the beam, and for a correction to be made for the energy not absorbed by the phantom.

The actual amount of energy absorbed in the phantom is expressed in ergs.

Division of the whole of the energy absorbed in the phantom (in ergs) by the integral of the ionization (the units employed for the ionization multiplied by grams) gives the value of the energy absorbed in ergs/g; i.e., the practical mean tissue dose.

If the units of the ionization are expressed as a percentage of the maximal ionization in the phantom, then the division of the absorbed energy by the integral of the ionization also gives the mean tissue dose in erg/g. If the ionization is expressed by the number of pairs of ions per cm^3 of the air space of the chamber, then division of the total energy absorbed by the ionization integral gives the product ρ_{air}.

To conclude, we will discuss the problem of the maximum allowable irradiation dose. Mayneord and Martin [2] calculated the flux of electromagnetic radiation in air—π, which constitutes a dose of 1 r for energies up to 100 Mev. If it is supposed that the maximum allowable dose of penetrating radiation and also of the normal γ-irradiation is equal to 0.05 r (and this is the value obtained from the results on the relative biological effectiveness), then from the results in [2] it is simple to derive the values of the maximum permissible daily energy fluxes (in ergs per cm^2 per day) and the maximum admissible intensities (in γ-quanta per cm^2 per second).

Multiplying the quantity π (in ergs per cm^2 per γ) by 0.05 gives the maximum allowable flux of 0.05 π (in ergs per cm^2 per day). By dividing this product by $6 \cdot 3600$* we obtain the maximum allowable intensity in ergs per cm^2 per second:

$$\frac{0{,}05\pi}{6 \cdot 3600} = 2{,}32 \cdot 10^{-6}\,\pi.$$

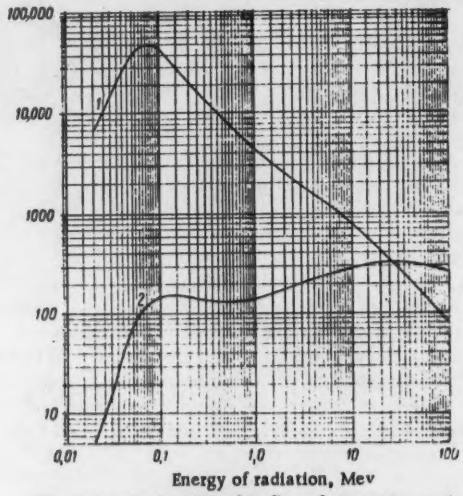

Fig. 6. Maximum allowable flux of electromagnetic radiation of different energies. 1—number of γ-quanta per cm^2 per second, per 0.05 r.e.p. per 6 hours; 2—incident energy in ergs per cm^2 per second per 0.05 r.e.p. per 6 hours.

When this figure is divided by the average energy of the γ-quanta, expressed in ergs, the maximum allowable intensity (in γ-quanta per cm^2 per second) is obtained, and this is:

$$\frac{2{,}32 \cdot 10^{-6}\pi}{1{,}6 \cdot 10^{-6}\,\bar{E}_\gamma} = 1{,}45\,\frac{\pi}{\bar{E}_\gamma},$$

where E_γ is the energy of the γ-quanta in Mev.

The values calculated in this way for the maximum allowable values of the different energy γ-quanta are shown in Fig. 6. They are very close to the values calculated by N. G. Gusev [4].

Received December 23, 1957.

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*6 • 3600 is the number of seconds in a 6-hour working day.

THE DISTRIBUTION OF IONIZATION ALONG A BEAM OF γ-RADIATION AND THE DETERMINATION OF RADIATION DOSE WITH NORMAL IONIZATION CHAMBERS

K. K. Aglintsev and G. P. Ostromukhova

It was established that the ionization effect, measured in normal ionization chambers for determining the dose in a region of γ-radiation, must be referred to a point in space, displaced by a distance of about $0.35\frac{R_{max}}{P}$ from the center of the measuring electrode of the chamber towards the source. The corrections for displacement of the measuring volume for the γ-radiation of Cs^{137} and Co^{60}, respectively, equal 0.996 and 0.992.

Normal ionization chambers (Fig. 1), operating with a high air pressure, are used for measuring radiation doses in roentgens in the energy region 0.3-3 Mev [1, 2]; the length of the measuring electrode is designated l_1. In chambers of this type, the measurement is reduced to determining the ionization current and the measuring volume 1.

The standard method of measuring the dose requires a knowledge of the point to which the value of the measured ionization effect refers. It is normally considered that the measured ionization current is proportional to the radiation energy absorbed in the measuring volume and that the value of the dose strength refers to the center of the measuring electrode.

Ionization in the chamber arises as a result of electrons, produced not only in the measuring volume, but also in volumes 2 and 3 in the beam, before and after the measuring volume.

The length l_2 between the planes perpendicular to the axis of the radiation beam, bounding volume 2, equals $\frac{R_{max}}{P}$, where R_{max} is the maximum range of electrons produced by the absorption of the measured γ-radiation in air at normal pressure, and P is the air pressure in the chamber. Correspondingly, l_3 is the distance between the planes perpendicular to the axis of the radiation beam, bounding volume 3. Ionization due to the absorption of radiation in volume 3 is caused exclusively by electrons undergoing back-scattering. Since back-scattering is very small in media with low atomic numbers [3], the contribution due to ionization in volume 3 will be insignificant and we can neglect it. Thus, it is sufficient to consider ionization due to the absorption of γ-rays in volumes 1 and 2.

In the present work it was established that the measured ionization effect should be referred to a point, displaced from the center of the measuring electrode towards the radiation source, and also, the value of this displacement was determined for Co^{60} and Cs^{137} γ-radiation. The calculations are based on a knowledge of the spatial and energy distribution of recoil electrons and a knowledge of the changes in ionizing capacity of electrons along their path.

The results of an experiment with Co^{60} γ-radiation confirmed the accuracy of the calculations.

Energy losses from electrons by ionization are not uniform along the path. Fig. 2 illustrates a graph constructed from Spencer's data [3], showing the energy loss ΔE for ionization by electrons with energies of 0.1-1 Mev in relation to the path length R, conditionally taken as unity. Practically 100% of the energy is lost by electrons along a run equal to 0.9 R.

Figure 3 illustrates the relative values of the ranges of electrons ejected at different angles, φ, during the absorption of Co^{60} γ-radiation in air, plotted in polar coordinates. Unity is taken as the range of electrons ejected at an angle of $\varphi=0°$, which equals $\frac{R_{max}}{P}$; for $t=20°$ C, $R_{max}=380$ cm.

For the calculation, each radius vector (Fig. 3) is divided into a number of sections, say 10, along which the given electron loses the same fraction of its original energy (10%). It is assumed that the energy is lost uniformly over the sections. The limits of corresponding sections on different radius vectors are joined by curves, and the areas bounded by the curves constructed and the zero radius vector are denoted by $S_1, S_2, \ldots S_{10}$. Each area formed, S_n (n has a value from 1 to 10 in Fig. 3), may in its turn be represented as the sum of elementary areas $S_{n\varphi}$, bounded by neighboring radius vectors (φ_1, φ_2). If all the areas $S_{n\varphi}$ are multiplied by the average number of electrons N_φ, produced in the interval of the angles φ_1, φ_2, then each of the sums of

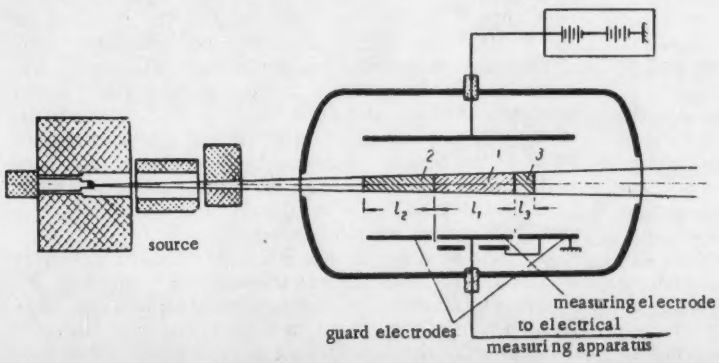

Fig. 1. Plan of a normal chamber for determining the dose in a region of γ-radiation.

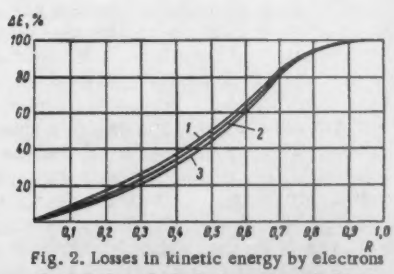

Fig. 2. Losses in kinetic energy by electrons along their path. Radiation energy in Mev: 1—0,5—0,7—1,0; 2—0,1; 3—0,3.

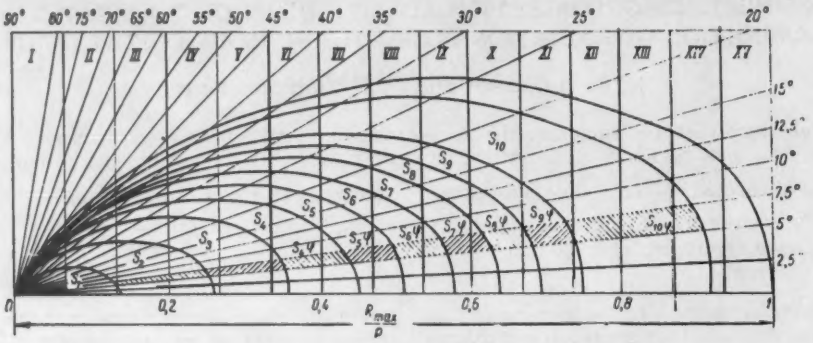

Fig. 3. Ranges of recoil electrons, ejected at different angles.

these derivatives $\sum_{\varphi=0^\circ}^{\varphi=90^\circ} S_{n\varphi} N_\varphi$ characterizes the energy loss, equal to 10% of the total energy E of all the recoil electrons. By dividing the zero radius vector into a series of equal sections by straight lines perpendicular to it, it is possible to determine the relative energy loss dE_m in ionization by all the recoil electrons for each section m (m has a value from I to XV in Fig. 3). By measuring all the elementary areas falling in the section considered with a planimeter and multiplying by we may determine the fraction of the corresponding sums $\sum_{\varphi=0^\circ}^{\varphi=90^\circ} S_{n\varphi} N_\varphi$ that they represent. The results for Co^{60} and Cs^{137} γ-radiation are presented in Fig. 4.

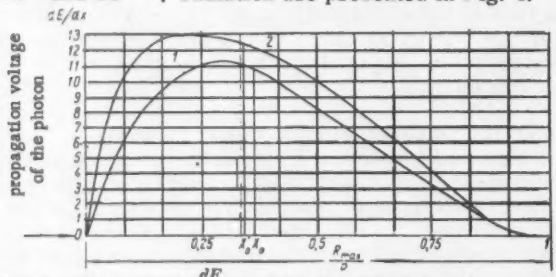

Fig. 4. Ionization losses $\frac{dE}{dx}$ along the direction of photon propagation for recoil electrons produced by the absorption of Co^{60} (curve 1) and Cs^{137} (curve 2) γ-quanta.

In Fig. 4, the areas bounded by the curves and the abscissa are proportional to the energy of the recoil electrons E.

Let us assume that a narrow beam passes through the ionization chamber with a radiation intensity $I_{\frac{l}{2}}$ in the center of the chamber. Let us divide the measuring electrode of the chamber into a series of elementary sections dx at a distance x from the end of the measuring electrode, where x varies from 0 to l. The ionization current in the elementary section dV, corresponding to the elementary section dx, is caused by electrons reaching this particular volume. The relative contributions of ionization from electrons arising at different distances from the volume will be different. The relative value of the ionization current in volume dV, produced by electrons recoiling from different centers of γ-quantum absorption, will change according to curves which are the mirror images of the curves in Fig. 4, reflected through a vertical.

Let us designate as x_0+x the coordinate of a point at a distance x_0 from the volume dV; x_0 is determined by the condition that the line through the point x_0 perpendicular to the direction of photon propagation divides the area bounded by the abscissa and curve 1 or 2 (Fig. 4) into two equal parts. Measurements showed that the values of x_0 for curves 1 and 2 were respectively:

$$x_0 = 0{,}36\,\frac{R_{max}}{P}\,; \qquad (1)$$

$$x_0' = 0{,}34\,\frac{R_{max}}{P}\,. \qquad (2)$$

The ionization current in the volume considered, dV, will be proportional to $I_{(x_0+x)}\, S_{(x_0+x)}\, dx\,(\tau+\sigma_\beta)$, where $S_{(x_0+x)}$ is the cross section of the beam with an average intensity $I_{(x_0+x)}$; $(\tau+\sigma_\beta)$ is the sum of the linear coefficient of photoelectric absorption and absorption with incoherent scattering.

By summing the values of the ionization current for all the elementary sections dx along the measuring electrode, we find that the measured ionization current in the chamber is proportional to the radiation energy absorbed in a volume displaced with respect to the measuring volume by a distance x_0, i.e. proportional to

$$I_{\left(x_0+\frac{l}{2}\right)} S_{\left(x_0+\frac{l}{2}\right)}\, l\,(\tau+\sigma_\beta),$$

where $S_{\left(x_0+\frac{l}{2}\right)}$ is the cross section of the beam at a distance x_0 from the center of the measuring electrode.

The calculated data for Co^{60} were checked experimentally. For this purpose a Plexiglas plate, 6 mm thick with a hole in the center was fixed in the ionization chamber. The plate was arranged between the radiation source and the measuring electrode, 25 cm long, in a direction perpendicular to the direction of photon propagation. The diameter of the hole, equal to 20 mm, was chosen so that the beam passing through cleared its walls. The ionization current produced by Co^{60} γ-radiation was measured for various distances L between the measuring electrode and the plate. The air pressure in the chamber, P, was 9.5 atm. The results of the measurements are given in Fig. 5 (curve 1). The value of the ionization current at $L=40$ cm equalled the value of the ionization current without the plate. When L was decreased to the minimum value (1 cm) the current in the chamber decreased.

The hole in the plate made it impossible to hold back all the electrons arising in the space in front of it. We measured the additional ionization arising from electrons passing through the hole from the space in front of the plate. By considering the diameter of the hole bounded by the plate and the steric distribution of

energy [4], carried in various directions by electrons arising from the absorption of Co^{60} γ-quanta, it was easy to determine the fraction of the measured ionization current J which constituted the additional ionization for various plate positions. On introducing into experimental curve 1 (Fig. 5) a correction for the electrons passing through the hole in the plate, we obtain

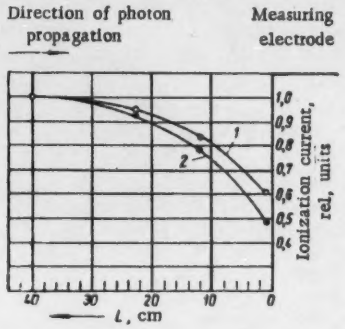

Fig. 5. Value of the measured ionization current in the chamber for different distances L between the plate and the electrode. 1—experimental curve; 2—corrected experimental curve.

curve 2. The abscissa of the point corresponding to 0.5 on the ordinate, and thus determining the position of point x_0, to which the measured effect should be referred, was found to be 1.5 cm. Consequently, the experimental determination gives the displacement of the point, to which the measured ionization should be referred, from the center of the measuring electrode towards the radiation source. This displacement equals 14 cm.

Calculation from equation (1) gives $x_0 = 14.5$ cm for the displacement; i.e., the results of calculation and experiment agree well.

From all that has been said, it follows that the generally accepted method of measuring γ-radiation dose with normal ionization chambers involves a systematic error due to an incorrect assumption on the absorption of γ-radiation in air. In order to remove this error, it is necessary to consider the absorption of radiation in a section x_0 between the effective point of γ-quantum absorption and the center of the measuring electrode; this is achieved by multiplying the measured value of the ionization current by the coefficient k, equal to

$$k = e^{-\mu P x_0},$$

where μ is the linear attenuation coefficient of the measured radiation in air at $P = 1$ atm and $t = 20°C$.

The numerical value of the correction coefficient k for Co^{60} equals 0.992 and 0.996 for Cs^{137}

Received September 1, 1958

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Letters to the Editor

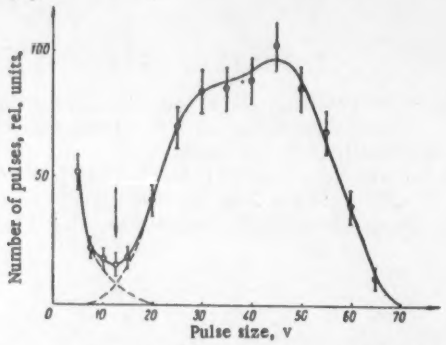
FISSION CROSS SECTION OF Am^{241} WITH 14.6 MEV NEUTRONS

A. N. Protopopov, Yu. A. Selitskii, and S. M. Solov'ev

Papers published previously [1, 2] give values of the fission cross section of Am^{241} for neutrons with energies of up to 7 Mev. In the present work we give the measurement of this cross section for neutrons with an energy of 14.6 Mev. The neutrons were produced by the reaction $T(d,n)He^4$.

The americium was deposited electrolytically on a platinum disc. A target of 15 mm diameter made from it was placed at a distance of 30 mm from the neutron source. The amount of americium was determined by the α-activity and the half life $\tau = 458.1 \pm .05$ years [3], and was 12 g. A check of the americium's purity on a magnetic α-spectrometer confirmed the absence from the target of any noticeable amount of fissile materials. The amount of Pu^{239} did not exceed 0.6%.

The procedure for determining the neutron flux falling on the target and the method of calculating the fission cross section were described in a previous paper [4].



Spectrum of Am^{241} fission fragments obtained with a one-channel analyzer with a channel width of 5v. The arrow marks the discriminator threshold at which an integral count of the fragments was carried out.

Due to the high α-activity of americium, a gas scintillation counter filled with xenon was used to register the fission fragments. the xenon ultraviolet radiation was converted to the visible range by terphenyl deposited on the inner walls of the counter. A test of the operation stability of such a counter showed that the terphenyl vapor evolved decreased the value of the light pulse by 10%. However, as the experiment for the cross section determination lasted not more than two hours, the stability could be considered satisfactory.

The light pulses from the counter were registered with an FEU-33 photomultiplier, which had a good resolving power and a large output current. After the multiplier, the pulses from the α-particles were intercepted by a fast-acting discriminator, built up on a DG-SZ crystal diode. The pulses passing the diode were shaped and after amplification entered a one-channel analyzer.

The figure shows a differential spectrum of Am^{241} fission fragments thus obtained. The low-value pulses were caused both by the superposition of pulses from α-particles and by nuclear reactions induced in the photomultiplier itself by the neutrons. The figure shows that the pulses from the fragments may be separated quite well from the background, and at the chosen discrimination threshold the amount of background pulses does not exceed 1% of the total number of fragments registered.

The fragment count was carried out with a statistical error of 2%. The neutron flux on the target was measured at the same time.

On the basis of data in paper [4], it may be taken that the fraction of Am^{241} fissions induced by scattered neutrons is negligible.

The fission cross section of Am^{241} for neutrons with an energy of 14.6 Mev was found to be equal to:

$$\sigma = 2{,}35 \pm 0{,}15 \text{ barns}.$$

The authors would like to thank G. I. Khlebnikova for preparing the target and V. G. Nedovesova for analyzing the target for purity on a magnetic α-spectrometer.

Received September 22, 1958

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AN INVESTIGATION OF THE DRAWING OF ELECTRONS INTO THE ACCELERATION SYSTEM OF BETATRONS AND SYNCHROTRONS

P. A. Ryazin and A. B. Minervin

For an exhaustive theoretical analysis of the problem of drawing electrons into the acceleration system in betatrons and synchrotrons, it is necessary to find a congruent solution to the multielectron problem, which takes into account the threshold and starting conditions. The starting equation for electron movement in a cylindrical system of coordinates r, θ, and z in a mean plane ($z=0$), without considering the effect of space charge, we will write in the form:

$$\frac{d}{dt}(m\dot{r}) = mr\dot{\theta}^2 - \frac{e}{c} r\dot{\theta} H_z, \tag{1}$$

$$\frac{d}{dt}\left(mr^2\dot{\theta} - \frac{e}{2\pi c}\Phi\right) = -\frac{e}{2\pi c^2} L \frac{dJ}{dt}. \tag{2}$$

Here e and m are the charge and mass of an electron (we will consider it variable); c is the velocity of light; $H_z = H_z(r, z, t)$ is the vertical component of the magnetic field strength; $\Phi = \int_0^r 2\pi r H_z\, dr$ is the magnetic

flux passing through a circle of radius r; L is the self-induction coefficient of the electron current circulating in the chamber and $J(t)$ is the electron current in the chamber to be determined. The right side of equation (2) is the moment of the self-induction force of the rotational electric field.

Let us introduce the following symbols: R_0 is the radius of an equilibrium betatron orbit, satisfying Videraux's condition; ρ is the instantaneous radial deviation of an electron from the equilibrium orbit; ρ_0 is the radial deviation of the momentary orbit from the equilibrium one; ρ_i is the distance from the injector filament to the equilibrium orbit; p is the distance from the filament to the edge of the injector; γ is the emergence angle of electrons from the injector; ω_0 is the Larmor frequency of an electron in the equilibrium orbit, and n_0 is the attenuation coefficient of the magnetic field

$$\left(H_z = H_0\left(\frac{R_0}{r}\right)^{n_0}\right).$$

The Coulomb interaction of the electrons of the beam appears as the effect of a quasi-static electric field, which varies in accordance with the changes in the nonstationary current in the chamber during the drawing-in. The beam, consisting of electrons with different "lifetimes," is, so to speak, coiled around the inner chamber walls near the equilibrium orbit. Let us consider that the net charge distribution forming in the chamber is uniform azimuthally and that the charge is cylindrical. The Coulomb reaction force has a basically radial direction. The field strength created by the charge $E_r = 2\pi q(\rho - \rho_c)$, where q is the charge density in the space filled by the current and ρ_c is the mean radius of the momentary electron orbits. At any moment of time

$$\rho_c = \frac{i_1\rho_{0_1} + i_2\rho_{0_2} + \dots}{J},$$

$$q = \frac{J}{\pi\rho^{*2}\omega_0 R_0},$$

where i_1, i_2, i_3 are the currents of separate turns, referred to the mean cross section of each turn; ρ_{0_1}, ρ_{0_2}, ρ_{0_3} are the radii of the momentary electron orbits for the same cross sections: $J = i_1 + i_2 + i_3 \dots$ and ρ^* is the radius of the toroid filled by the electron charge. Then the radial force acting on an electron

$$Fr = eE_r = \frac{2Ie(\rho - \rho_c)}{\rho^{*2}\omega_0 R_0}.$$

In all the formulas given, L and are in CGSM units, the current is in CGSE units and r and p are in cm.

Instead of n_0 one should introduce the dynamic attenuation coefficient of the magnetic field $n(t) = n_0 + \Delta_n$ where

$$\Delta n = \frac{2e}{m\rho^{*2}\omega_0^2 R_0}.$$

The relativistic effect of the attraction of the moving charges may be neglected due to the low energy acquired by the electrons during the drawing-in. We will also assume that the functions $J(t)$, $\omega_0(t)$, $n(t)$ and $m(t)$ change slowly in comparison with the harmonic functions, which describe betatron oscillations adjacent to the momentary orbits. Then, the equations of motion (1) and (2) are integrated, Coulomb interaction forces introduced, the result linearized, and a series of transpositions carried out, using van der Pol's method, to give the following final form:

$$\rho(\tau, \tau_0) = \frac{\varkappa\tau_0 \frac{1+\delta\tau_0}{1+\delta\tau} - \varepsilon_0 J(\tau)\rho_c(\tau)}{1-\varepsilon_0 J(\tau)} - \frac{\varkappa}{2}[J(\tau) - J(\tau_0)] + \sqrt{\frac{1+\delta\tau_0}{1+\delta\tau}}\sqrt{\frac{1-\varepsilon_0 J(\tau_0)}{1-\varepsilon_0 J(\tau)}}\left\{(\rho_i - \rho_{0\max})\times\right.$$
$$\times\cos 2\pi\sqrt{1-n_0}\int_{\tau_0}^{\tau}(1+\delta\tau)\sqrt{1-\varepsilon_0 J(\tau)}\,d\tau +$$
$$+\frac{R_0\gamma}{\sqrt{1-n_0}}\times\frac{1}{\sqrt{1-\varepsilon_0 J(\tau_0)}}\sin 2\pi\sqrt{1-n_0}\times$$
$$\left.\times\int_{\tau_0}^{\tau}(1+\delta\tau)\sqrt{1-\varepsilon_0 J(\tau)}\,d\tau\right\}, \qquad (3)$$

$$0 = 2\pi\int_{\tau_0}^{\tau}\left\{1+\delta\tau - \frac{\rho_0}{R_0} + \frac{\rho_{0\max}}{R_0} - \frac{n_0\rho_{0\max}}{R_0}\right.$$
$$+\varepsilon_0 J(\tau_0)(1-n_0)\frac{\rho_c(\tau_0)-\rho_{0\max}}{R_0} -$$
$$\left.-\frac{L[J(\tau)-J(\tau_0)]}{2V_0 T_0(R_0,0)}\right\}d\tau. \qquad (4)$$

In these equations $\rho_0 = \frac{\varkappa\tau_0 - \varepsilon_0 J(\tau_0)\rho_c(\tau_0)}{1-\varepsilon_0 J(\tau_0)}$, $\tau = \frac{t}{T_0(R_0,0)}$ dimensionless time [$T_0(R_0, 0)$ is the time of one revolution of an equilibrium electron];

$$\varkappa = \frac{R_0 T_0(R_0,0)\mu}{1-n(\tau_0)}\left[\frac{\Delta V}{2V_0} - \frac{2\pi 10^{-8} F H_{\max}}{H_0}\right]$$

is the change in the starting radii of the momentary orbit during one revolution of an equilibrium particle (ΔV is the change in the injection voltage per μsec.); V_0 is the equilibrium injection voltage; H_0 is the equilibrium strength of the magnetic field; $H_{\max}$ is the amplitudinal value of the magnetic field strength at the betatron orbit;

$$\delta = \frac{\Delta H}{H_0(R_0,0)} = \frac{H_{\max}\,2\pi F T_0(R_0,0)}{H_0(R_0,0)}$$

is the relative change in the magnetic field strength in one revolution of a particle;

$$\varepsilon_0 = \frac{6\cdot 10^9 e}{m\rho^{*2}\omega_0^2 R_0(1-n)}$$

is a parameter which allows for the effect of the Coulomb interaction and $\mu = \frac{LR_0}{2V_0(1-n_0)[1-\varepsilon_0 J(\tau)]T_0}$ is the contraction or expansion of an orbit in cm, relative to a 1 amp increase in current. In formula (3) and in the following ones, L is in henries, $J(t)$ is in amp and the lengths are in cm. τ and τ_0 are calculated from the moment that the starting orbit coincides with the equilibrium one.

The distribution of current density with respect to the angles at which the electrons emerge from the injector is taken to be the equilibrium one. Consideration of this fact and the threshhold conditions of equations (3) and (4) make it possible to determine unequivocally the coordinates of the electrons and currents in the chamber. Using the iteration method, one can calculate with great accuracy the pulses of the nonstationary current in the chamber and find all the main characteristics of the betatron or synchrotron. When $J = 0$, equations of the one-electron theory are obtained, which are applicable in the case of low emission currents. With large emission currents, the determining factor is the mechanism of collective interac-

tion. Drawing-in occurs both at the front and the back edge of the pulse. However, there are qualitative differences between these two cases.

Drawing-in at the front edge. At the moment when the beam "breaks" (the separation of the head of the current from the source) there is strong asymmetry of the peripheral momentary orbits relative to the average beam orbit. When the current falls after the beam break, there is strong contraction of the orbit (in millimeters per revolution), which removes the tail section of the beam from the dangerous zone, and as a result this part of the beam is drawn into the accelerating system (first term in equation (3)). The greater maximum current in the chamber with respect to the moment of beam break, the stronger this effect. The induction mechanism of drawing-in (second term in equation (3)) promotes an increase in the current in the chamber before the break in the beam and thus intensifies the action of the Coulomb effect. In this case the electron beam is similar to an elastic ring.

Drawing-in at the rear edge. When the beam breaks at the inner wall, the beam is approximately symmetrical relative to the threshold orbit. Here, when the current decreases after the break in the beam, the Coulomb effect is seen as a sharp decrease in the amplitudes of the betatron oscillations (third term in equation (3)) and as a result, the tail section of the beam passes the dangerous zone and is drawn into the acceleration. This effect has a stronger action, the greater the current in the chamber at the moment of the break in the beam. In this case also, the induction mechanism promotes an increase in the current before the break in the beam.

Drawing-in at the flat section of the pulse is essentially the same as at the rear edge.

With the use of an internal injector, drawing-in at the front edge is similar to drawing-in at the rear edge with the use of an external injector and vice versa. The Coulomb interaction eliminates critical drawing-in relative to the value n.

With small emission currents, as a result of the increase in the magnetic field, the displacement of the orbit is a micron per revolution, which, as the authors showed, is still sufficient for drawing-in as part of the electrons pass the injector many times (15 – 20 times).

A detailed account of the theory and of the methods for accurate and approximate calculation of betatron characteristics is being prepared for publication.

Received February 10, 1958

ISOLATION OF Np^{239} IN A RADIOCHEMICALLY PURE STATE BY THE USE OF THE NUCLEAR RECOIL OF FISSION PRODUCTS

Yu. A. Zolotov and I. P. Alimarin

It is advantageous to use the nuclear recoil of fission products to separate Np^{239} from the fission products. Fragment nuclei with high energies have a comparatively high range in the mother material (e.g. 8 μ in U_3O_8 [1]); on the other hand, the range of U^{239} and Np^{239} formed by neutron capture and β-decay is insignificant. By using sections of the mother substance with dimensions smaller than the range of the fragments, it is possible to free the mother substance from the activity of fission products, leaving in the uranium the activity caused by the presence of only Np^{239} (after the decay of U^{239}), UX_1 and UX_2.

Decay curve of Np^{239}

Joliot [2], Wolfgang [1] and Henry and Herczeg [3] used the recoil of nuclear fragments to extract active fission products in a pure form. Wolfgang also paid attention to the possibility of a practical separation of actinides in a radiochemically pure state. We carried out experiments which are a continuation of Wolfgang's work.

For the irradiation we used uranium dioxide in which the U^{235} content had been reduced to 0.4%. The use of impoverished uranium for the preparation of Np^{239} considerably reduced the total fission product activity.

The filling material used was calcium carbonate. The particles of uranium dioxide were distributed uniformly in the mass of $CaCO_3$ by precipitating the latter from a saturated solution of calcium nitrate, in which the particles of UO_2 had first been dispersed. The voluminous precipitate formed trapped the grains of UO_2. After being dried, the precipitate was irradiated in a reactor. The filler was separated from the UO_2 by dissolving the $CaCO_3$ in hydrochloric acid, with subsequent centrifuging and washing of the uranium dioxide, which was then dissolved in nitric acid.

An extraction method was used for the final purification of Np^{239} from uranium, UX_1 and UX_2. After an ether extraction of the oxidized neptunium together with the uranium, the neptunium was extracted from the ether solution by action of reducers, which convert it into the unextractable pentavalent state. Hydrazine (cold) was found to be an efficient reducer for the reextraction of neptunium.

Procedure. Uranium dioxide was ground in an agate mortar and by means of sedimentation fractions were removed, which consisted of particles precipitat-

ing at a rate of $\leqslant 2{,}16 \cdot 10^{-3}$ cm/sec. According to Stokes' law, the average diameter of the particles taken was about 2 μ. The concentration of the suspension was determined by a gravimetric method, and then an aliquot part of it was transferred to a beaker, diluted and saturated with calcium nitrate. After careful mixing, calcium carbonate was precipitated with a concentrated solution of ammonium carbonate and the precipitate filtered, washed, dried and weighed. The weight ratio UO_2: $CaCO_3$ varied from 1 : 100 to 1 : 500 in different experiments.

The mixture obtained was irradiated in a nuclear reactor (neutron flux, $7 \cdot 10^{12}$—$2 \cdot 10^{13}$ neutr./cm^2 · sec) for 12 – 48 hours. The irradiated sample was dissolved in the minimal amount of cold 1.5 N HCl, the solution centrifuged and the precipitate washed 2 – 3 times with 1.5 N hydrochloric acid and 3 times with water.

The washed uranium dioxide was dissolved in a small (about 2 ml) amount of hot, concentrated nitric acid, the solution partially diluted and potassium bromate added to it to oxidize the neptunium to the hexavalent state. The solution (about 0.1 M with respect to $KBrO_3$) was then heated for 15 min at 90 – 100°C. Aluminum nitrate* was added to the cooled solution, which was diluted (to 1.5 M with respect to $Al(NO_3)_3$ and 1 M with respect to HNO_3) and transferred to a separating funnel for extraction with ether. The combined ether extracts (after 3 – 4 extractions) were treated with a reducing solution for reextraction of the neptunium (0.4 M with respect to HNO_3, 8 M in NH_4NO_3 and 0.1 M in hydrazine hydrochloride).

Results. The purity of the Np^{239} isolated was checked by the half-life; the decay curve did not show deviations from linearity over 24 days (see figure).

The method described makes it possible to separate NO^{239} in 1 – 2 hours, with the activity of the fission products removed in the first minutes. The mixture of uranium oxide and filler particles may be prepared beforehand and used over a long working period.

The possible partial adsorption of fission products on the particles of uranium oxide, which apparently occurs during solution of the filler [4, 5], is not a serious drawback, since the small amount of fission products is separated from the NO^{239} during the extraction and reextraction.

Besides $CaCO_3$, it is possible to use as fillers other substances, which can be kept in the high temperature of the active zone of a reactor during a long irradiation period. Considering this, we do not consider the irradiation of aqueous suspensions of uranium oxides, stabilized with gelatin [1, 3], as practically convenient.

Received March 14, 1958.

*Ammonium and calcium nitrates were also used for salting out.

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THE EFFECT OF RADIATION ON THE VALENCE STATE OF PLUTONIUM IN PERCHLORIC ACID SOLUTIONS

N. I. Popov

In a previous article [1] a report was given on the results of the action of x-radiation on the valence state of plutonium in nitric acid solutions. The oxidation of tetravalent plutonium observed during the irradiation was explained by the action of hydroxyl radicals. However, this interpretation was not conclusive, since no answer was found to the problem of the effect on plutonium of short-lived unstable oxides of nitrogen, whose formation under the action of radiation is quite possible.

To eliminate this effect it is necessary to use an anion which is insensitive to the action of radiation. The ClO_4^- ion was chosen. As is known, it is not completely inert to the action of radiation, but it has been established experimentally that the behavior of plutonium under the action of x-radiation does not depend on the concentration of the ClO_4^- ion in solution.

We divided the different valence states of plutonium into only two kinds: oxidized and reduced. However, instead of the lanthanum sulfate method of separating these forms of plutonium, we used the lanthanum fluoride method, which is well known in the literature. To avoid side reactions we did not use "hold-back oxidants," considered necessary in paper [2].

RESULTS AND DISCUSSION

Solutions of plutonium in perchloric acid. Figure 1 shows the results of irradiating solutions of plutonium in 0.3 N $HClO_4$. As in [1] the total concentration of

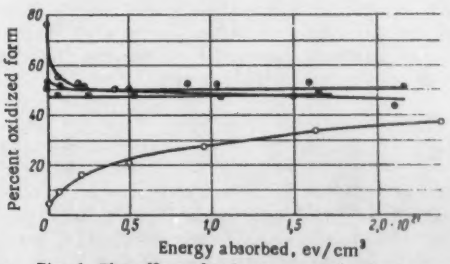

Fig. 1. The effect of radiation on the valence state of plutonium in solutions.
Total concentration of plutonium, $3.6 \cdot 10^{-4}$ mole/liter. ○ and ● 0.3 N $HClO_4$; ■ – 0.3 N $HClO_4$ + 2.0 M $NaClO_4$.

plutonium was about 10^{-4} mole/liter. As the figure shows, the plutonium in these solutions may be reduced or oxidized under the effect of radiation, depending on the ratio of its oxidized and reduced forms in the original solution.

This ratio tends to a certain stationary value. If the concentrations of the two forms of plutonium before irradiation correspond to the stationary ratio of these forms during the action of radiation, then no obvious change in the valence forms of plutonium is produced during irradiation.

Increasing the concentration of the ClO_4^- ion in the solution from 0.3 to 2.3 M did not have any effect on the radiation behavior of plutonium. This may be considered as an indication that the radiolysis products of the ClO_4^- ion do not interact with plutonium ions (at least, they do not oxidize Pu^{+4} and do not reduce PuO_2^+).

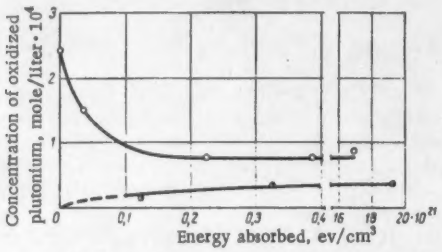

Fig. 2. Change in the valence state of plutonium under the action of radiation in a solution of 0.3 N $HClO_4$ + 0.03 M $UO_2(ClO_4)_2$, containing various ratios of oxidized and reduced forms of plutonium. Total concentration of plutonium, $3.4 \cdot 10^{-4}$ mole/liter.

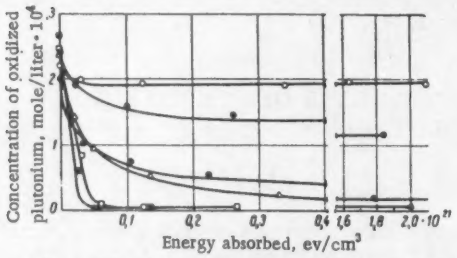

Fig. 3. The action of radiation on the valence state of plutonium on 0.3 N $HClO_4$ solutions, containing $UO_2(ClO_4)_2$ (mole/liter): ○) 0.001; ◑) 0.01; ●) 0.06; △) 0.10; □) 0.20; ■) 0.30. Total concentration of plutonium, $3.4 \cdot 10^{-4}$ mole/liter

The data obtained indicate that the change in the valence state of plutonium in aqueous solutions is caused by the action of radiation through the radicals H and OH in the absence of acceptors reacting with them, as was proposed in paper [1].

Solutions of plutonium together with $UO_2(ClO_4)_2$. Both the reduction and oxidation of plutonium occur in these solutions during irradiation and a stationary ration of oxidized and reduced forms is established [at moderate concentrations of $UO_2(ClO_4)_2$]. This ratio depends on the concentration of $UO_2(ClO_4)_2$ in solution, but it may vary for the same concentration of $UO_2(ClO_4)_2$, depending on the side from which it is approached (Fig. 2).

If the concentration of the oxidized form of plutonium in the original solution is greater than in the stationary

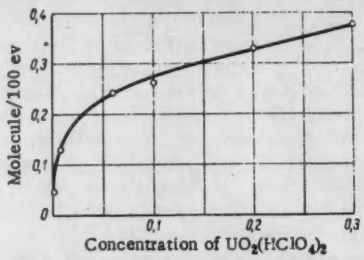

Fig. 4. Dependence of initial plutonium reduction yield on $UO_2(ClO_4)_2$ concentration in 0.3 N $HClO_4$ solutions. (For an initial oxidized plutonium concentration of $2.4 \cdot 10^{-4}$ mole/liter).

state reached during irradiation, then irradiation produces reduction of plutonium. In contrast to nitric acid solutions containing $UO_2(NO_3)_2$, here the reduction process begins in the first moment of irradiation, at any $UO_2(ClO_4)_2$ concentration (Fig. 3).

The initial reaction yield and the degree of reduction increase with a rise in the $UO_2(ClO_4)_2$ concentration. At concentrations of the latter exceeding 0.3 M the reduction occurs so rapidly that it was impossible to determine the yield under the experimental conditions.

In these solutions reduction goes to completion. The corresponding experimental data are presented in the table.

The Reduction of Plutonium by the Action of Radiation in 0.3 N $HClO_4$ Solutions in Relation to the $UO_2(ClO_4)_2$ Concentration

$UO_2(ClO_4)_2$ concentration, mole/liter	Initial concentration of oxidized plutonium, mole/liter	Initial reduction yield, ions/100 ev	Ratio Pu_{ox}/Pu_{red} in the stationary state
0,001	$2,2 \cdot 10^{-4}$	0,042	1.0
0,01	$2,4 \cdot 10^{-4}$	0,13	0.54
0,03	$2,3 \cdot 10^{-4}$	0,19	0.20
0,06	$2,0 \cdot 10^{-4}$	0,20	0,04
0,1	$2.4 \cdot 10^{-4}$	0,26	0,02
0,2	$2,4 \cdot 10^{-4}$	0,33	0
0,3	$2.7 \cdot 10^{-4}$	0,42	0

These results agree with the premise put forward previously, that, for the ratios of uranyl and plutonyl ions investigated, the reduction of plutonium under the action of radiation in the presence of UO_2^{++} ions is not due to atomic hydrogen, but ions of pentavalent uranium (UO_2^+).

At sufficiently high uranyl salt concentrations, when the formation of UO_2^+ proceeds most efficiently, the disproportionation of these ions becomes noticeable:

$$2U(V) \rightleftarrows U(IV) + U(VI).$$

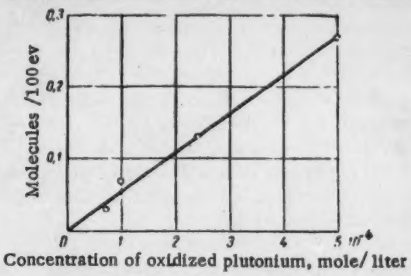

Fig. 5. Relation of initial plutonium reduction yield to the concentration of oxidized plutonium in a solution of 0.3 N $HClO_4$ + 0.01 M $UO_2(ClO_4)_2$.

The disproportionation of pentavalent uranium apparently decreases the plutonium reduction yield with an increase in the $UO_2(ClO_4)_2$ concentration (Fig. 4).

Figure 4 was constructed from data in the table, presented for the same concentration of oxidized plutonium, equal to $2.4 \cdot 10^{-4}$ mole/liter. It was found that the plutonium reduction yield in the solutions considered depends on its concentration also.

The corresponding relation for a solution containing 0.01 M $UO_2(ClO_4)_2$ is presented in Fig. 5. In the concentration region examined the reduction yield is directly proportional to the concentration of oxidized plutonium with its total content constant.

Received February 6, 1958

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DETERMINATION OF THE SPECIFIC α-ACTIVITY AND THE HALF-LIFE OF U^{233}

Ya. P. Dokuchaev and I. S. Osipov

The specific α-activity of U^{233} was determined on five samples of uranium, whose isotopic compositions, determined from mass spectrometric analysis data, varied between the limits (% by weight):

U^{232}*	$\sim 4 \cdot 10^{-3}$
U^{233}	93,28–95,24
U^{234}	2,25–2,36(±0,07)
U^{235}	<0,1
U^{238}	2,46–4,38(±0,07)

One nitric acid solution was prepared from each sample by a gravimetric method. In calculating the titers of the solutions, allowance was made for 0.15 by % weight of total inactive impurities. Thirty preparations were made from each titrated solution. The preparations were measured on two pulse ionization chambers with a small solid angle. The 4 π counting factor was 2963 ± 2 for one chamber and 2944 ± 2 for the other. In calculating the counting factors, the α-activity was considered as distributed in a circle 10 + 0.5 mm in diameter.

*The content was determined from the energy of the α-particles.

Corrections were not introduced for counting error or absorption in the preparation itself, as the count rate was within the limits of the resolving power of the apparatus and the active layer of the preparation was measured for 180 min (90 min on each chamber).

From determinations of the α-activity of five samples, we found that one microgram of U^{233} corresponded to an average of 20,940 α-decays/min. If we consider that all the samples contained about 0.05% by weight of the practically inactive (in comparison with U^{233}) isotope U^{235}, then the specific α-activity of U^{233} should be considered as 20,950 ± 100 α-decays/min per μg.

Thus, the specific α-activity of U^{233} corresponds to a half-life of $(16.26 \pm 0.08) \cdot 10^4$ years. Within the limits of accuracy this half-life value agrees with literature data [1, 2].

Received September 20, 1958

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DETERMINATION OF THE SPECIFIC α-ACTIVITY OF Pu^{239} AND Pu^{240}

Ya. P. Dokuchaev

The specific α-activity of Pu^{239} and Pu^{240} was determinted on twelve samples of plutonium, whose isotopic compositions varied between the following limits (% by weight):

Pu^{239}	91,26–99,11
Pu^{240}	(0,87±0,06)–(7,57±0,1)
Pu^{241}	$(1,3 \pm 0,3) \cdot 10^{-2}$–$(1,04 \pm 0,05)$
Pu^{238}	$(1,62 \pm 0,1) \cdot 10^{-3}$–$(1,75 \pm 0,05) \cdot 10^{-2}$
Pu^{242}+Np^{237}	0,01–0,1

One titrated solution was prepared from each plutonium sample by a gravimetric method. In calculating the titer it was considered that the total of the inactive impurities in all the samples was 0.15 – 0.20 % by weight.

Twenty-five preparations were made from each solution by a gravimetric method. The activity of the preparations after drying and firing was measured in parallel in two chambers with small solid angles. The

4π counting factor was 18,804 ± 15 for one chamber and 18,861 ± 15 for the other. In calculating the counting factors it was considered that the α-active substance was distributed in a circle 10 ± 0.5 mm in diameter.

No correction was introduced for counting error, as the count rate did not exceed 450 counts/min. The resolving power and the "plateau" of the apparatuses were checked carefully. Each preparation was measured for 90 min on each chamber (a total of 180 min). The specific α-activity of $Pu^{239} + Pu^{240}$ was determined from the difference of the measured activity and the activity of $Pu^{238} + Am^{241} + Pu^{241}$.

The number of α-decays from $Pu^{238} + Am^{241}$ was determined from the energy of the α-particles, and the number of α-decays constituting the contribution of Pu^{241} was calculated from the decay scheme [1 – 3]

$$Pu^{241} \xrightarrow[13 \text{ years}]{\beta} Am \xrightarrow[470 \text{ years}]{\alpha}$$
$$\alpha \downarrow 3 \cdot 10^{-3}\%.$$

The correction for the contribution from $Pu^{242} + Np^{237}$ did not exceed 1 – 2 α-decays/min.

These data were processed by the method of least squares and gave the following result for the calculated specific α-activity of $Pu^{239} + Pu^{240}$:

1 μg of Pu^{239} gives 136,200±200 α-decays/min, which corresponds to a half-life of 24,390 ± 30 years;

1 μg of Pu^{240} gives 500,000 ± 4000 α-decays/min, which corresponds to a half-life of 6620 ± 50 years.

The errors are given for a reliability of 0.6745 from the Student – Fisher table.

The half-life values obtained for $Pu^{239} + Pu^{240}$ agree completely with the most accurate values reported in [4] and [5].

Received September 20, 1958

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BURN-OUT HEAT FLUX FOR FLOW OF WATER IN PIPES

I. T. Alad'ev, L. D. Dodonov, and V. S. Udalov

The burn-out heat flux q_{cr} appears to be the most important characteristic of heat transfer during boiling. There are no analytical methods of determining its value, and the experimental results are very limited. The burn-out flux has been determined in [1-10] for the heat transfer to water flowing in channels which is subcooled below the saturation termperature t_s. In most of these works q_{cr} was measured for relatively small variations in the parameters defining the regime, particularly in the pressure. The results of experiments carried out in the heat-transfer laboratories of the Energy Institute of the Academy of Sciences USSR, in 1956-1957 are given below.

The experimental setup consisted of a chrome-nickel steel tube in a closed loop (Fig. 1). Circulation of water in the loop was maintained by a rotary pump. Pressure was raised and maintained by a steam pressurizing vessel, the contents of which also served as a source of supply for the loop. The deaeration of the water was carried out in an expansion tank. The desired temperature of the water in the loop was established and maintained by using a cooler and an auxiliary heater. The experimental tubes used were seamless and thin-walled (δ = 0.4 mm) and made of 1Kh18N9T steel, of internal diameter d = 8.2 mm and length from 35 to 133 mm.

During the experiments the following were measured: the pressure, the temperature and flow of the water, the temperature of the wall. The pressure in the loop was measured downstream from the experimental section, using a portable manometer of a type with accuracy 0.35. The flow of water through the experimental section was measured by thermocouples set in sleeves upstream and downstream from the experimental tube. The temperature t_j of the water at the outlet of the tube was used to calculate the subcooling.

The heat flux was calculated from the strength of the current and the electrical resistance of the tube, and was controlled through the voltage drop along the tube. The variations in these measurements did not exceed 5%. The current was determined from the fall in voltage across a shunt (2000 amp/45 mv, of accuracy class 0.5) and the resistance of the tube at a

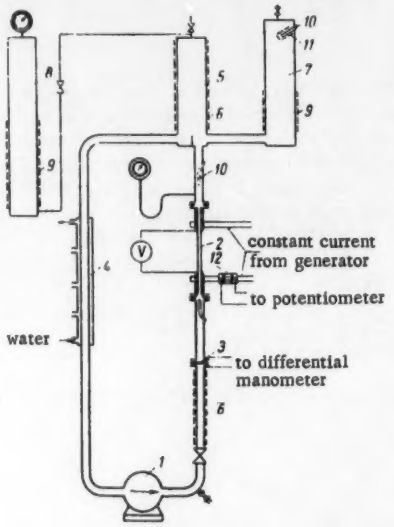

Fig. 1. General arrangement of the experimental setup:

1) Circulating pump; 2) experimental tube; 3) flowmeter orifice; 4) cooler; 5) expansion tank; 6) auxiliary heater; 7) small pressurizing vessel; 8) large pressurizing vessel; 9) pressurizing vessel heaters; 10) thermocouple sleeve; 11) screen; 12) 2000 amp/45 mv shunt.

temperature equal to the arithmetic mean of the temperatures of the outer and the inner surfaces of the wall. The outer surface temperature was measured by a thermocouple with the junction fixed to the tube by glass filaments at the middle section, and the inner surface temperature was determined by calculation. The dependence of the electrical resistance of the tube on temperature was found experimentally by comparison with a standard resistance.

The temperature of the water and of the walls of the tube, the saturation temperature, and also the voltage drop across the shunt at the time of the experiment were recorded by a EPP-09 self-recording electronic potentiometer of class 0.5, the scale of which was previously calibrated. The readings of the potentiometer permitted the determination of the value of the burn-out flux and the temperatures of the water and the wall (at the mid-section) at burn-out. The errors of measurement in single results were the following: for q, 3-5%; for w, up to 3%; and for Δt_H, not more than 2°C.

Before the experiments the water was deaerated by boiling in the apparatus at atmospheric pressure for 30-40 min. The pressure, the flow of water, and the heat flux were then brought to their desired values. (All these operations were carried out at relatively low temperatures.) Critical states were then brought about slowly by raising the temperature of the water (not more than 1 deg per min) with constant p, w, and q. The onset of burn-out was determined in two ways. In the majority of experiments it was done visually, by watching for the reddening of the tube, using an angle mirror to observe the whole surface. For $p = 20$ and 40 atmos, burn-out was also determined with the help

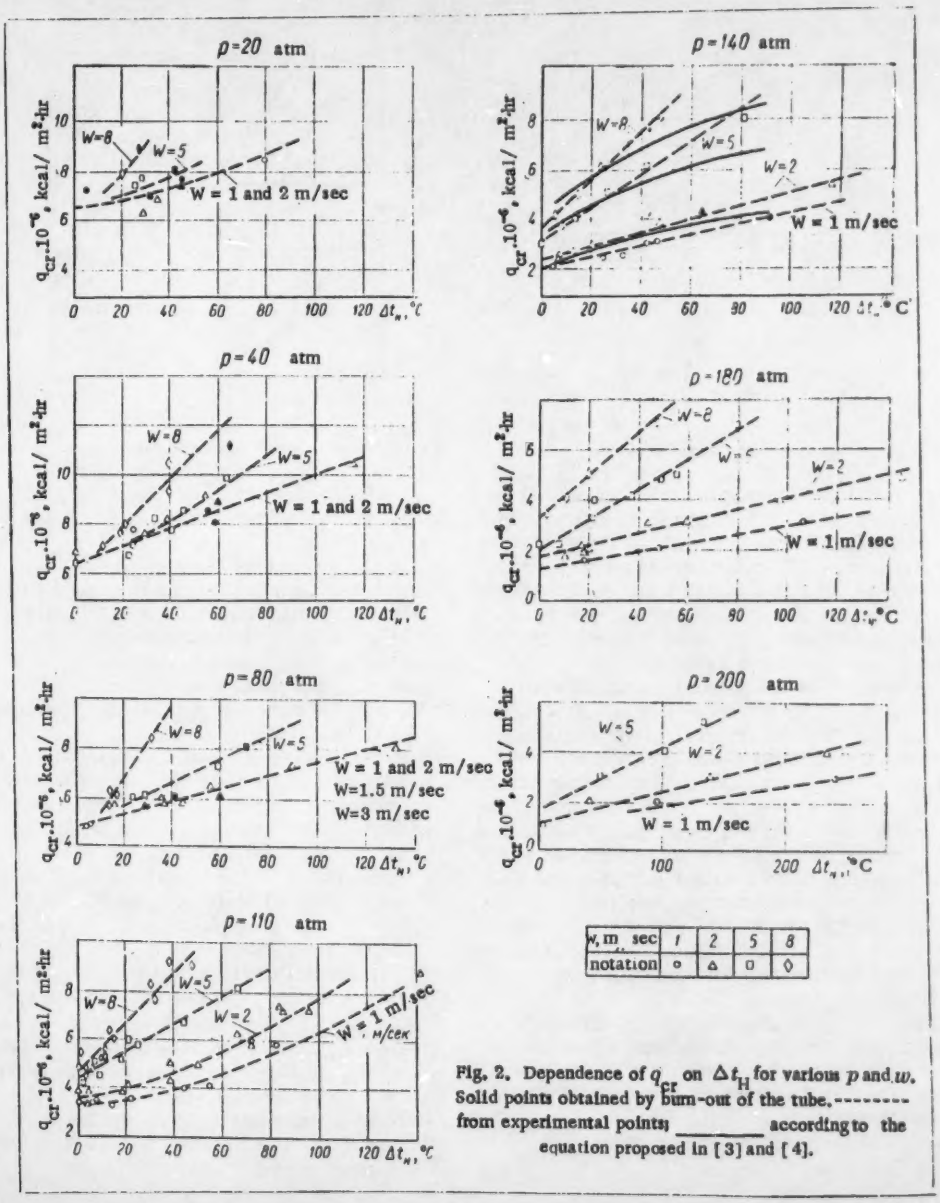

Fig. 2. Dependence of q_{cr} on Δt_H for various p and w. Solid points obtained by burn-out of the tube. -------- from experimental points; _____ according to the equation proposed in [3] and [4].

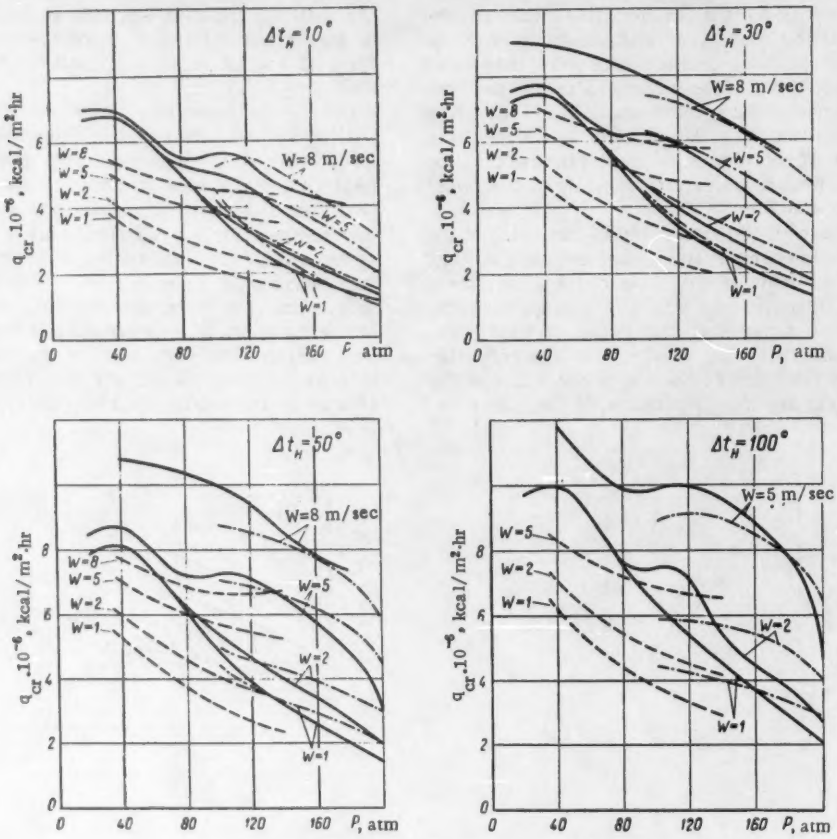

Fig. 3. Comparison of results of various investigations.
——results of the authors; —·—·— results in [9]; — — results in [5] and [6].

of two differential thermocouples or one hyperthermocouple connected with an electronic regulator which switched off the load at the onset of burn-out.

For low pressures (20 and 40 atmos) and water velocities of the order of 1 to 2 m/sec burn-out developed rapidly, and as a rule the tube burned out. In these experiments significant ocillations (pulses) were noticed in the pressure and the velocity.

For pressures above 140 atmos the critical state developed more slowly: a red spot of diameter 2-4 mm appeared on the tube (at three to five millimeters from the upper end) and this spot grew rather slowly, so that there was a good chance of switching off the current and thereby preventing the destruction of the tube. The value of q_{cr} for the experiments was taken to be the flux corresponding to the appearance of the red spot on the tube or to the destruction of the tube.

The experiments were carried out in series for loop pressures p of 21, 41, 81, 111, 181, and 201 atmos, and water velocities of 1, 2, 5, and 8 m/sec. In each series q_{cr} was determined for constant p and 2 and for several Δt_H.*

The values of q_{cr} obtained in the experiments are shown graphically in Fig. 2. Values of w and Δt_H correspond to conditions at the outlet of the experimental tube. The results given in 3) and 4) are plotted on the same graph.

$\Delta t_H = t_s - t_j$, the rise in temperature of the liquid up to saturation

On analyzing these results it can be concluded that:

a) the dependence of q_{cr} on p, w, and Δt_H is complicated;

b) if p increases from 40 to 300 atmos (other conditions remaining the same) q_{cr} decreases; q_{cr} is practically the same for $p = 20$ and $p = 40$ atmos*;

c) Δt_H and q_{cr} increase together and the influence of Δt_H on q_{cr} is greater for larger w;

d) if w increases then q_{cr} also increases; for $p = 20$, 40, and 80 atmos and $\Delta t_H = 20°$ the influence of w is almost unnoticeable.

In Fig. 3 is shown the relation $q_{cr} = f(p)$ for a series of values of w and Δt_H. The continuous curve approximates the mean of q_{cr} derived from the graphs of Fig. 2. The results of other authors are also shown (see [5], [6]† and [8], [9]). The curves representing the result of [8] and [9] were drawn by using the dimensionless-parameter equation of [9]‡.

It can be seen from Fig. 2 and Fig. 3 that the difference between the results of this paper and those of [8] and [9] (there are more details on the range of variation of the physical and geometrical parameters

*Values of q_{cr} obtained by the author for $p = 1$ atmos. $w = 2$ m/sec and various values of Δt_H were significantly smaller than the values for $p = 29$ atmos. for the same w and Δt_H.

†Carried out with a tube of diameter 5.7 mm.

‡Values of q_{cr} calculated from the formula recommended in [8] are low; in some cases the difference exceeds 25%.

in [9] exceed 25% only in isolated cases. The difference between the results of [3] and [4] and our results is also generally less than 25%. From these considerations it can be concluded that the values of the burn-out flux and their dependence on the parameters of the setup are reliable for the conditions investigated.

From Fig. 3 it can also be concluded that the character of the dependence of q_{cr} on p, w, and Δt_H in [5] and [6] is the same as in our work; however in [5] and [6] the values of q_{cr} themselves are distinctly smaller, in single cases as much as 50%. It is difficult to decide the cause of these differences since the details of the experiments in [5] and [6] have not been published.

It follows from the formula proposed in [5] and [6] for calculating q_{cr} that the burn-out fluxes for a tube of diameter 3.7 mm will be greater than for a tube of diameter 5.7 mm. We carried out three experiments (p = 40 and 80 atmos, w = 8 m/sec) with a tube of diameter 3 mm. The values of q_{cr} obtained in these experiments were much higher than those in experiments with a tube of 8.2 mm. The influence of the dimensions of the channel on the values of q_{cr} for low pressures was also established in [7]. In contradistinction to this it was concluded in [8] and [9] that for $p \geq 100$ atmos, q_{cr} was independent of the diameter of the tube (for $4 \leq d \leq 12$ mm). The values of q_{cr} obtained for p = 180-210 atmos in channels formed from tubes with a minimum distance between them of 1 mm were, according to [10], the same as for a single tube.

When all these results are considered it can be concluded that the shape and dimensions of the channels within certain limits have an influence on the values of q_{cr}. It is evident that this influence is related to the size of the steam bubbles, and it is therefore not possible to extend the results of [8-10] to the case of lower pressures, smaller dimensions, or lower liquid velocities.

No comparison has been made with the experimental results of [1], [2], or [7], where the pressures were low (up to 22 atmos) and channels with shapes (slots, ring-shaped gaps) and dimensions different from those in the present paper were used. It should be mentioned, however, not only that the pressure appeared to have no influence on q_{cr}, but also that the values of q_{cr} calculated using the proposed formula differ essentially among themselves (by 50-200%).

From an analysis of the work presented here and of that in the papers mentioned, the following conclusions may be drawn:

a) Values of the burn-out flux for water subcooled below the saturation temperature flowing in a tube ($d \geq 8$ mm) and in channels of other configurations (with a gap $h \geq 8$ mm) can be reliably determined from the results of the present work for pressures of 20 to 200 atmos, and for $p \geq 100$ atmos from the results of [8] and [9]. The differences between our results and those of [3], [4] (p = 140 atmos), [8], and [9] do not exceed 25%.

b) The shape and dimensions (diameter, gap) of the channel within definite limits have an influence on the values of q_{cr}. The possibility of extending the conclusion of [8-10], that q_{cr} is independent of the dimensions of the channel, to cases different from those considered in these investigations needs verification.

c) In [7] no influence of the pressure on q_{cr} was observed for p = 1-21 atmos. This result raises serious doubts.

Received November 3, 1958

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THE INSTITUTE OF NUCLEAR PHYSICS OF THE ACADEMY OF SCIENCE, UZBEK SSR

The Institute of Nuclear Physics of the Academy of Science of the Uzbek SSR was created in 1956. The purpose of the Institute is to investigate problems connected with peaceful applications of atomic energy and, in particular, fundamental problems in nuclear physics.

The work at the Institute concerns, on one hand, the application of radioactive isotopes and nuclear radiations in various fields of industry for the purpose of automation in a number of processes and the development of new apparatus for investigations in the field of nuclear physics. On the other hand, it concerns investigation of the effect of radiations on solids and liquids, particularly the effect of gamma rays on silk, cotton and cotton seed.

One group of scientists at the Institute, namely S. A. Azimov, M. Yu. Borukhov and B. K. Mal'tsev, has developed an automaton for the determination of the position of cable splices. This apparatus is installed at the "Tashkent Cable" plant.

An automatic regulator of liquid levels has also been developed. Its construction is simpler and more reliable than the "UR-4" type of level-measuring apparatus. The laboratory model of the regulator has already been tested in the laboratory and the apparatus will be installed at the Chirchiks Chemical Combine. This regulator was developed by M. Yu. Borukhov and D. N. Normetov.

Work is in progress on the automation of a two-step cycle for pulverization and classification of sulfide ores for the Altyn-Topkan Polymetal Combine; the designers are S. A. Azimov, M. Yu. Borukhov and A. L. Lebedev. Further, a unification regulator is being developed for equipment whose functioning can be regulated with radioactive isotopes.

Another group works on the construction of an apparatus for determining the density and humidity of soil at great depths, using the absorption and scattering of gamma radiations of radioactive isotopes.

An improved method of measuring density by the use of gamma rays has been developed at the Institute. The error of measurements was decreased by 1.5%.

Together with the Institute of Water Problems and Hydrotechnology of the Academy of Science of the Uzbek SSR the Institute of Nuclear Physics works on the solution of a number of practical problems connected with reclamation of the "Hungry Steppes." Among these are problems of dynamics of underground waters, control of the settling of the sides of canals, and diffusion of water and salt solutions through porous materials.

At the radiogeology laboratory work is being done on the application of activation analysis for determining the concentration of dispersed elements in mountain deposits.

The scientists of the laboratory for the application of nuclear radiations to the study of the physics of metals are developing a stereoscopic gamma ray apparatus for detecting defects in metals. The laboratory gives practical help to a number of Tashkent organizations.

All this work is performed at the Technology Section of the Institute of Nuclear Physics.

In the Nuclear Physics Section investigations are being carried out on neutron physics, nuclear reactions, electronics, mass-spectroscopy, radiation physics and chemistry; preparations are being made for use of the equipment constructed at the Institute for Research. Among the apparatus are a research reactor of the VVR type, a cyclotron, accelerating tubes and a powerful gamma ray apparatus. At present a cobalt apparatus with a source activity of 4000 curies is used for research in radiation physics and chemistry. In particular, together with the Institute of Silk Industry and the Physicotechnical Institute of the Academy of Science of the Uzbek SSR, investigations are being made on the effect of gamma rays on the preservation and destruction of cocoons of the moraceae silkworm. It has been shown that the pupa can be killed by gamma radiation; the lethal dosage is being determined. Also, the effect of gamma rays on the mechanical properties and resistance of raw silk is being determined. Analagous investigations are being carried out with cotton fiber. The work on the destruction of pupae was done by U. A. Arifov, G. A. Klein, G. A. Gumansky, S. Z. Pahinski, and others.

Investigations are being made on the effect of gamma rays on the absorption properties of solids, the electrical conductivity of semiconductors, on aqueous solutions of carbohydrates, etc.

After the reactor is in operation at the Institute work on neutron physics, nuclear reactions and activation analysis will develop. The nuclear reactor will also be used to obtain radioactive isotopes and, first of all, the short-half-life isotopes necessary for scientific research institutes and industries of the Republic of Middle Asia.

The Institute is a center which provides methods and technical help in the application of radioactive isotopes and nuclear radiations to the scientific research organizations and industrial organizations of the Republic.

METALLOGRAPHY OF PLUTONIUM

Because of its unique properties (6 allotropic modifications, negative expansion coefficient of the δ-phase, etc.) metallic plutonium has a great interest for metallographers. Data on the technique of plutonium metallography were published recently [1].

Aside from the necessity of taking special precautions because of the radioactivity of plutonium [2], which requires the work's being done in isolated chambers and buildings with forced air ventilation [3], the pyrophoric property of some of its alloys must be taken into account.

The initial samples are kept in double-walled plastic containers; large samples are kept in refrigerators to slow down oxidation. Pyrophoric samples are sealed in evacuated capsules made of high-melting-point glass and the capsules are kept in double-walled plastic con-

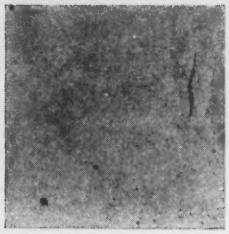


Fig. 1. Cast α-plutonium obtained by zone melting.

Fig. 2. β-plutonium. The alloy contains 10 atomic % of uranium. One can see the Pu_2O_3 inclusions (dark) and probably carbides enriched in uranium (×500).

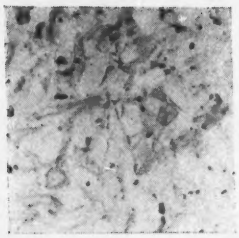


Fig. 3. Cast δ-plutonium. The alloy contains 9 atomic % of cerium. The interdendritic components are not identified (×250).

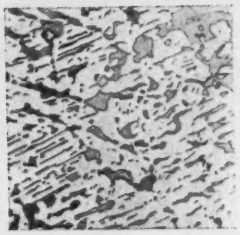


Fig. 4. The ξ-phase in α-uranium obtained as the result of eutectoid decomposition of β-uranium containing 15 atomic % of plutonium, at 500°C (×500).

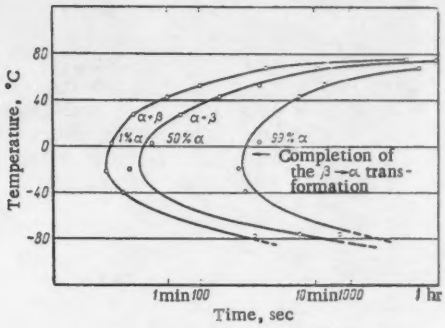
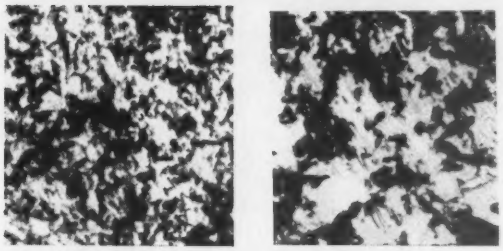


Fig. 5. Time-temperature-transformation curves for the $\beta \rightarrow \alpha$ transformation. The sample containing 0.05% impurities was first homogenized for one hour at 175°C. (×300).



−21° C −73° C

Fig. 6. Effect of hardening temperature for the transformation $\beta \rightarrow \alpha$ on the grain size. Etching solution: 2 volumes tetraphosphoric acid; 2 volumes water; and 5 volumes of ethylene glycol. Polarized light (×300).

tainers inside metal jars. The metal wastes and contaminated materials and reagents are also kept and transported in plastic containers inside metal jars.

Samples prepared for polishing are usually mounted in plastic (methylmetacrylite). Heating above the temperature of the first α—β transformation (122°C) must be avoided. Polishing is done with sandpaper and finished with aluminum oxide in carbon tetrachloride, or with diamond dust (1 micron) at low speeds (125-250 rpm).

Since no satisfactory chemical agent is yet known, the etching is done electrolytically with reagents based on tetraphosphoric ($H_6P_4O_{13}$) or orthophosphoric acids, of the following compositions: 7 volumes of tetraphosphoric acid, 36 volumes of water and 57 volumes of ethylene glycol; or, 8 volumes of orthophosphoric acid, 5 volumes of glycerin and 5 volumes of ethyl alcohol (this reagent is also used for uranium).

The first reagent puts in evidence the microstructures of the α- and β-plutonium and the mixtures of these allotropic modifications (Figs. 1 and 2).

Samples of about 0.5 cm^3 are etched in a small volume of electrolyte at room temperature with a voltage of 5-12 volts — the process is controlled more conveniently by the voltage than by the intensity of current. The action of the reagent can be weakened by changing with proportion of the components of the electrolyte to the ratio 12:33:55. This ratio is good for etching the δ-phase (Figure 3). For slow etching of soft allotropic modifications the best results are obtained with the 2:3:5 proportion and a voltage of 12 volts. If 26 volts are maintained for 3 minutes one obtains an anodized surface, optically active in polarized light, which reveals the grains of the metal; so far, there is no etching method to make the grain boundaries of plutonium visible in the light field.

The second reagent is useful for etching plutonium and its alloys with 5-10 volts (Fig. 4).

The duration of electrolytic etching is 10-45 seconds; it is prolonged only when it is necessary to produce an anodized surface or to remove a deformed layer. For lower voltages (within the limits indicated for each reagent) the etching is stronger; for higher voltages electropolishing is predominant. Plutonium is passivated after electrolytic etching and the sample can then be washed in distilled water and dried in warm air.

Etching can also be done by ionic bombardment in argon under a pressure of 35 microns of mercury (with 2000 volts at 12 ma for 30 minutes).

Mechanically polished samples are "dirty" and should be examined only in sleeve chambers. After etching, the samples can be examined almost normally. The samples are transported with forceps, which are kept in special containers near the microscope. The microscope, the microhardness tester, etc., are provided with removable plastic tops (covers) which are thrown out as they become contaminated. The time of safe manipulation outside the sleeve chamber varies (according to the composition and rate of formation of loose oxide on the surface) from a few minutes to 8 hours. One finds [4,5] interesting data on the kinetics of α—β-, β—γ- and γ—β-transformations of plutonium, which occur at temperatures below 200°C with considerable changes in volume: 8.9% for the α/β transformation and 2.4% for the β/γ transformation. The time-temperature-transformation curves given in one of the publications [4] are typical of isothermic reactions in eutectic alloys (Fig. 5).

From these results the following conclusions can be drawn:

1. The maximum rate of the β—α transformation occurs at a temperature of about -20°C; the rate depends on the amount of impurities in plutonium; at this temperature 50% of the plutonium containing 0.02% impurities is transformed in 20 seconds, while it takes 70 seconds to transform 50% of the plutonium containing 0.15% impurities. The lower the transformation temperature the smaller the grain size of the metal (Figure 6).

2. The α—β transformation begins below 123.5°C and becomes very rapid at temperatures above 135°C.

3. The maximum rate of γ—β transformation occurs at 95°C. When the temperature is lowered (to about 80°C) the formation of the α-phase begins. The rate of α phase formation is higher than the rate of the β—α transformation even at those temperatures at which the γ-phase is transformed into β-plutonium. Thus at -78°C the γ—α transformation takes less than 10 seconds.

4. Increase of the temperature of preliminary homogenizing annealing somewhat reduces the rate of γ—β and β—α transformations.

After several cycles of heat treatment $\alpha \rightleftarrows \beta \rightleftarrows \gamma$ plutonium becomes porous.

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BRIEF COMMUNICATIONS

USSR. Work is nearing completion on the construc- of the first research reactor in the Transcaucasus, under the auspices of the Institute of Physics of the Academy of Sciences of the Georgian SSR. Erection of the central physical hall has been completed. Assembly of the reactor, control panel and consoles is underway. Laboratories are equipped to carry out research experiments in different branches of nuclear physics, chemistry, biology and geology.

Britain. On November 9, 1958, the first of four reactors at the atomic power station is Chapelcross went critical. At the present time, experimental work is in progress on preparing the reactor for full-power operation (the electric power delivery of the plant, with one reactor operative, is 45 Mw). All of the reactors should be in operation by the end of 1959.

Britain. General Electric Co. has reported development of a new process for producing graphite which is completely impermeable to gases, as well as techniques for graphite-graphite bonding and graphite-metal bonding. The graphite may find application as cladding for high-temperature fuel elements cooled by carbon dioxide, hydrogen, etc. A tube fabricated from this graphite was first seen on display at the Second International Conference on the Peaceful Uses of Atomic Energy at Geneva. A pressure of 10 atm of carbon dioxide was maintained inside a tube heated to 600°C, with a vacuum retained outside the tube.

Britain. Experimental coils needed for investigation of the field inside a magnet pole piece assembled from steel laminae have been manufactured for the 7 Bev proton synchrotron at Harwell. The coils, fabricated from aluminum tubing (35 x 22 mm^2) with an inner diameter of 12.5 mm, consist of four pairs of layers each having 20 turns of coil. The dimensions of the coil are 1.8 x 1.5 m^2, the weight 2.8 tons.

Britain. Metropolitan Vickers Co., Ltd. has recently designed an oil diffusion pump known as the "Metrovac 9/14." This pump lays the basis for the high vacuum system of the ZETA facility. There are two variants of the pump: an oil pump and a mercury pump, with power outputs of 2.5 kw and 3 kw, respectively. Peak pumping speed for the oil pumps, at pressures of 10^{-4} mm Hg, is 2,800 liters/sec for air and 3,000 liters/sec for hydrogen. The critical temperature, for a trap temperature of 15°C, is less than 10^{-5} mm Hg for apiezon oil G and less than 10^{-7} mm Hg for apiezon oil C. At a trap temperature of –40°C, this pressure is less than $5 \cdot 10^{-8}$ mm Hg. The maximum back pressure is 0.3 mm Hg for apiezon oil G and 0.11 mm Hg for apiezon oil C. Peak pumping speed for the mercury pumps comes to 2,500 liters/sec for air and 8,000 liters/sec for hydrogen. The peak back pressure is then 0.15 mm Hg.

Britain. The Rochester division of the Consolidated Electrodynamics Corporation (CEC) has landed an order from Brookhaven National Laboratory in the USA for the fabrication of high-vacuum pumps for the proposed 30 Bev proton synchrotron (48 ion-evaporation and 12 mechanical booster pumps). CEC will in addition deliver 20 ion-evaporation pumps for the 50 Mev linear accelerator which will function as the proton injector for the synchrotron.

An ion-evaporation pump is a device for obtaining a high vacuum without the use of organic fluids. In this case contamination of the vacuum chamber, due to reverse flow of the evaporator in oil diffusion and mercury diffusion pumps, is averted.

Britain. On December 8, 1958, the last of the four reactors installed at the atomic electric power plant at Calder Hall achieved criticality.

Britain. The Atomic Energy Control Board has decided to build an experimental boiling water reactor at Windscale or at Dounreay. This decision was adopted under the impetus of the successful operation of the ANL experimental boiling water reactor EBWR.

Britain. The National Institute for Nuclear Research has signed a contract for the manufacture of electrical equipment to service the power supplies of the electromagnet assembly in the 7 Bev proton synchrotron now under construction at Harwell.

Belgium. A Belgian firm has been awarded a EURATOM contract for the manufacture of 408 coils for a 25 Bev proton synchrotron.

India. India's second reactor, named "Zerlina," designed for the study of new fuel-element designs, will be started up in mid-1959.

Italy. A 1.1 Bev electron synchrotron will be put into operation during the spring of 1959, at Frascati.

Italy. Ground-breaking ceremonies were held on November 20, 1958 for the atomic electric power plant to be built at Latina. The plant construction is contracted to a British firm.

Italy. Construction of an atomic power plant delivering 150 Mw electric power (the ENSI project) has been decided upon, the site to be the town of Punta Fiume at the mouth of the Garigliano river (near Naples). The plant will be an improved two-cycle facility based on a water boiling reactor. The power plant will be similar to the one now being built at Dresden (Ohio) USA. In contrast to the Dresden power station, project ENSI has two loops in the secondary cycle of each turbogenerator. About 70% of the equipment was supplied by Italian companies. Construction work on the power plant will begin this year and is scheduled for completion in early 1963.

Canada. The NRX reactor at the Chalk River research center was shut down due to an accident. A fuel element ruptured while being replaced and part of it, being still in in-pile location, burned up. The fire was quickly put out, but the reactor building suffered contamination by radioactive dust. The NRX reactor was built in 1947. It went through a first accident in December 1952, as a result of which the reactor was shut down until February 1954.

USA. The first international symposium on fuel elements for nuclear reactors was held January 28-29 at Columbia University. The symposium was held under joint auspices of Sylvania-Corning Nuclear Corporation and Columbia University.

USA. An experimental heterogeneous water boiling reactor is being built at Greendale, Wisconsin (light-water moderation and normal pressure). The reactor core consists of two concentric areas: the outer region contains fuel elements made of aluminum-clad uranium dioxide; the inner region, acting as a steam generator, consists of tubular fuel elements, the fuel of which is a cermet made from stainless steel and uranium oxide; the cladding is of stainless. It is ex-

pected that the construction of the reactor should be finished in June or July 1959.

USA. Pacific Gas and Electric Co. is planning to build an atomic electric power plant delivering 60 Mw of electric power near the town of Eureka, California. The new plant, similar to the one built at Shippingport, will cost 20 million, which comes to 300 per kw, compared to 1,200 per kw for the Shippingport station. The cost of the power delivered will be 0.8 cent per kilowatt hour, comparable to the cost of electric power obtained from a conventional power plant located in the same part of the country.

USA. On October 21, 1958, at Idaho Falls, the first of two reactors for land-based prototypes of prime movers for two atomic warships now under construction (the cruiser USS Long Beach and the carrier Enterprise) went critical.

USA. At Cambridge, Massachusetts, a conference on charged-particle accelerators, organized on the initiative of the High Voltage Engineering Corp., met from November 14 to 16, 1958. The renowned physicist R. J. Van de Graaff of MIT took part in the proceedings.

USA. W. Zinn, AEC chairman, announced the findings of an investigation into an accident at Oak Ridge, caused by a critical excess as an aqueous solution of UO_2F_2 was being poured into a cylindrical vessel 762 mm in diameter. The critical height (about 127 mm) was exceeded. The resulting liberation of energy was so powerful that part of the solution was ejected and caused damage to the bottom end of the cylinder.

USA. An electric motor capable of operation 1,000 hrs at a stretch at red-hot temperatures was displayed at Pittsburgh. The refractory properties of the novel motor owe to the use of heat insulation material which is capable of withstanding extremely high temperatures, as well as to the use of graphite motor bearings.

It is proposed that a motor of this type may prove feasible in nuclear power applications.

USA. A developmental technique for winning uranium ores by diamond drilling of bore holes is under study. The technique consists in leaching disperse uranium impregnations with chemicals introduced via the bore holes. Laboratory tests of the technique have yielded satisfactory results.

France. In late August 1958, the Atomic Energy Commissariat announced that the 2.5 Bev proton synchrotron at the Saclay nuclear research center, known as "Saturn," was brought into operation. The diameter of the annular magnet of the accelerator measures 17 meters across, the peak magnetic field intensity reaches 13.5 kilogauss, the number of particles per pulse is 10^7–10^8. The cost of the accelerator facility runs to 6 billion francs.

West Germany. Construction work has begun on a 15 Mw atomic electric power station near Frankfurt, for the Rheinisch Westphalisch firm. A boiling water reactor using fuel elements slightly enriched with uranium dioxide will be installed in the plant. The steam output of the plant will be 105 ton/hr, and the steam pressure will be 45 atm.

West Germany. At Gestacht, near Hamburg, the fourth swimming-pool reactor in the country, a 5 Mw research pile for seagoing prototype tests, was started up. The reactor was built by the American outfit Babcock and Wilcox Co.

West Germany. Construction was initiated in Hamburg, in October 1958, on a 6-8 Bev electron synchrotron. Laboratory facilities are being built for the nuclear physics division of the State Physical Institute. The accelerator, expected to cost 60 billion marks, will begin operation no earlier than 1962.

Information and Bibliography

STANDARDS AND DEFINITIONS

Research reactor. A reactor designed for carrying out experiments and investigations with the aid of neutrons and gamma rays. Research reactors exist in different power levels, reactor types and purposes. They are of the utmost importance, since advanced work in the utilization of atomic energy is unthinkable without the performing of experiments involving the use of research reactors. In recent years, not only physical research, but also research associated with other branches of science has been carried on using these reactors: materials testing, chemistry, medicine, biology, agriculture, etc. The term "experimental" is often applied to this reactor type in the literature, stressing the use of the reactor for conducting experiments. However, the term "research" gives a fuller and more rounded reflection of the purpose of the reactor.

Experimental Reactor. A reactor whose design is viewed as experimental. To test out the design, the reactor is operated for a certain limited time, while its behavior in operation is observed carefully with a view towards eliminating defects before producing industrial reactors on a larger scale. All power-only nuclear reactors now in operation in the USSR, USA and UK are in essence experimental reactors, since their creation and operation pursue the aim of testing one or other selected type or design over an extended period of time, and choosing, for further development and refinement, the most economical type or the most reliable operational design among them. The term "experimental reactor," contrasted to the term "research reactor," reflects amply the purpose of such a reactor as of developmental design.

Swimming-Pool Reactor. A reactor whose core is situated (suspended, usually) in a pool of light water. The water is in this case fulfills the role of moderator, tamper, coolant and, to an appreciable extent, shielding. Thanks to simplicity of design and low costs, reactors of this type have met with wide favor as research reactors. In the literature, reactors of this design have been termed occasionally "submerged reactors," "reactors with pour-off shielding (s nalivnoi zashchitoi)", "aquarium reactors." Although these terms accurately reflect the basic design of such reactors, their use may lead to confusion and ambiguity. The term swimming-pool reactor (basseinovyi reaktor) characterizes more fully the basic concept underlying the reactor type.

NEW LITERATURE

Atomic Engineering Abroad, No. 1, 1959. (Atomnaya Tekhnika za Rubezhom)

This issue contains review articles on the peaceful uses of thermonuclear energy (E. Teller), on the development of accelerator work in the USA (K.R. Symon),

physics of the fission process (R.B. Leachman), and also a number of articles on nuclear power engineering. Brief remarks on the articles included in this issue are appended below.

E. Teller. Peaceful Uses of Thermonuclear Fission. Paper No. 2410 (USA). Second International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958.

The paper gives a brief summary of the problems met in attempting to harness thermonuclear reactions, and gives a short review of the basic papers submitted by American scientists in that field.

K.R. Symon. Development of Accelerators in the United States. Paper No. 2400 (USA).

As companions to the existing large accelerators (with particle energies of the order of several billion electron volts), several new accelerator facilities, based on the latest advances in accelerator engineering, are in the design or construction stage in the United States.

The University of Pennsylvania and Princeton University are building a weak-focusing proton synchrotron at 3 Bev particle energy and a time-averaged current level equal to 0.5 microamps (100 times higher than that obtained at the Brookhaven cosmotron).

A strong-focusing 6 Bev electron synchrotron is being built by several organizations headed up by MIT, and will have a pulse rate of up to 60 pulses per sec.

At Argonne National Laboratory, design of a 12.5 Bev proton synchrotron using zero-gradient magnets has been completed. The zero gradient allows extensive savings in magnet weight.

Construction has begun at Brookhaven on the 25 Bev alternating-gradient proton synchrotron.

A linear accelerator measuring 3.2 km in length is being developed at Stanford University, and is expected to accelerate electrons up to energies of 15-45 Bev.

Research is being conducted in the USA to create fixed-field alternating-gradient focusing accelerators yielding sizable energy increments and increases in beam intensity. A 15 Bev proton accelerator is being designed to produce head-on collisions between protons (to obtain the equivalent interaction of a beam with a fixed target in a conventional accelerator, the protons would have to be accelerated up to 540 Bev).

R.B. Leachman. Fission. New Data and Concepts. Report No. 2467 (USA).

This is a review throwing light on the most recent advances in the field of fission mechanism studies in neutron-induced fissions. Attempts to find an explanation for the asymmetry of the distribution of fragments with respect to mass are discussed briefly, in addition to the special features of light-nuclei fission and the role of fission energy.

V.A. Elliott, R.D. Maxon, W.D. Nixon and J.W. Marriman. The Dresden Atomic Power Station. Report No. 2372 (USA).

A description is given of the design of the 180 Mw Dresden atomic-fired electric power station, which will be on the line in mid-1960. The station reactor is of the boiling water type, moderated and cooled by light water, and 1.5% enriched uranium dioxide sinter is used for fuel. The energy yield is 10,000 Mw-day per ton.

The processes of fuel charging, achieving criticality and startup are dealt with in detail. Double-cycle steam generation and other improvements cut electric power costs to 0.75 cent per kilowatt, which is competitive with the cost of electric power generated by the most advanced heat power plants.

H.G. Davey, J. Butron and T.N. Marsham. Operating Experience at the Calder Hall Power Plant. Report No. 1522 (UK).

By March 1958, 690 billion kilowatt-hours of electric power had been produced at the Calder Hall plant. The article, on the basis of operational experience accumulated, gives characteristics of the performance and recommendations for the improvement of assemblies and systems such as the startup and control system, the system interconnecting the electrical and reactor facilities, the fuel recharge system, the system for detecting faulty fuel elements and the coolant loop.

Nuclear Power Engineering in West Germany (abstract). Report No. 1806 (West Germany).

Taking as point of departure the growing power demands evident, West Germany has outlined an extensive program for the building of large power reactors, including Calder Hall types, heavy water reactors, pressurized-water reactors and high-temperature gas-cooled reactors. Orders will be placed for construction projects in 1961. It is expected that the proposed power stations will be in operation in 1965.

The abstract reports the presence in West Germany of a series of research reactors, and reviews some of the practical problems associated with carrying through a program of nuclear power development.

R.B. Holden, M.C. Kells, C.I. Whitman. Manufacture of Metal Beryllium Based on the Continuous Electrolytic Process. Report No. 717 (USA).

The authors reveal that it is possible to cheapen the cost of beryllium by using a manufacturing process proposed by them, which yields a beryllium amalgam through continuous electrolysis of a mixture of fused $NaCl$-$BeCl_2$ in a mercury cathode. The article provides a description of the method for manufacturing the metal, accessory equipment and the properties of the amalgam. The metal obtained by the method proposed lends itself to direct hot pressing.

L.E. Link et al. The "Mighty Mouse" High-Flux Research Reactor. Report No. 423 (USA).

The article gives a description of the physics and power engineering incorporated in the American "Mighty Mouse" reactor, as well as its design, control system, charging and test facilities. At a power level of 250 Mw in the test zone, the neutron flux will exceed 10^{15} neutrons $\cdot$ cm^{-2} $\cdot$ sec^{-1}.

Grozdenskii, D. E. Radiobiology. Published by "Zdanie," 1958, 32 pages, 60 kopeks.

This brochure expounds, in popular form, the physical-chemical fundamentals of the action of ionizing radiations on living organisms. The author succeeds in informing the reader, in a concise and at the same time in a not oversimplified manner, of the nature and properties of ionizing radiations and on the processes which take place in living tissues exposed to radiation. The author uses, in order to explain the mechanism involved in the biological response to radiation, the latest achievements in domestic and foreign science. The book will prove to be of value to an audience having only a general scientific background (on the level of 9th-10th year middle school) and desirous of improving their knowledge of radiobiology fundamentals.

Ermakov, G.V. Powerful Atomic Electric Power Stations. Published by "Zdanie," 1958, 32 pages, 60 kopeks.

This brochure describes two atomic power stations now being built in the USSR: a 400,000 kw plant with reactors of the reactor type used in the first Soviet Atomic electric power plant, and one of 420,000 kw power using water-cooled water-moderated reactors. Information provided by the author on the design of the power stations has been published previously. The pathways of nuclear power development abroad and power reactors being built abroad are described briefly. The prochure is written in very popular style.

Bagnall, K.W. Chemistry of the Rare Radioelements. Polonium-Actinium. Butterworth's Scientific Publications, London, 1957, 177 pages.

In this monograph, unique of its kind, we find assembled a variety of physical-chemical data on such interesting elements are polonium-84, astatine-85, radon-86, francium-87, radium-88 and actinium-89.

The author devotes a good deal of attention to experimental techniques for dealing with trace quantities of polonium, radium and actinium. Much of the book is given over to a description of methods of polonium processing.

The book is written for an audience of scientific research workers and students, but may prove also useful to a wider readership interested in learning about the chemistry of the rare elements.

Cahen, G., Treille, P. Precis d'Energie Nucleaire. Preface by F. Perrin. (Introduction to Nuclear Power). Paris, Dunod, 1958, 356 pages.

The book summarized the scientific fundamentals of industrial nuclear power use. The book contains 9 chapters: the beginnings of modern physics; elements of nuclear physics; nuclear reactors; general data on reactors; nuclear radiation hazards; shielding; fissile and structural reactor materials; radiation dosimetry; industrial use of nuclear power; industrial use of radiations.

Grainger, L. Uranium and Thorium. G. Newnes, Ltd. London, 1958, 204 pages.

The book is a monograph describing the properties, technology and chemistry of uranium and thorium, based on the latest available data, as well as different methods for extracting those elements from ores, winning them in the form of metals, and a review of the areas of application of those elements and their future perspectives.

Hot Laboratory Equipment. Sup't. of Documents, US Government Printing Office, Washington 25, D.C.

A manual for scientists, engineers and supervisory personnel in atomic industrial enterprises. It contains detailed material and design data on equipment and accessories for working with moderate and large amounts of radioactive materials.

Hughes, D.J., Schwartz, R.B. Neutron Cross Sections. BNL-325, 2nd ed., BNL, 1958, 325 pages.

A second fully revised edition of the widely renowned atlas of neutron cross sections, prepared by Brookhaven National Laboratory in the USA, has appeared in print. The first edition of the atlas was timed for the first International Conference on the Peaceful Uses of Atomic Energy (Geneva, 1955). Since that time, it has become the standard manual on neutron cross sections. The second edition of the atlas was prepared for the beginning of the second Geneva conference. The atlas contains carefully selected cross-section values for thermal neutrons and resonance parameters, and well as curves of partial and total cross sections as functions of neutron energy. The atlas presents more up to date data obtained in various laboratories over the world. It will be of interest to both reactor specialists and nuclear physicists.

Nuclear Handbook. G. Newnes, Ltd., London, 1958, 624 pages.

In the handbook, compiled by a team of specialists (O. R. Frisch acting as consultant), multifarious data necessary in the everyday work of research scientists conducting their studies in the field of atomic energy are made available.

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AN SSSR	*Academy of Sciences, USSR*
FIAN	*Physics Institute, Academy of Sciences USSR*
GITI	*State Scientific and Technical Press*
GITTL	*State Press for Technical and Theoretical Literature*
GOI	*State Optical Institute*
GONTI	*State United Scientific and Technical Press*
Gosenergoizdat	*State Power Press*
Gosfizkhimizdat	*State Physical Chemistry Press*
Gozkhimizdat	*State Chemistry Press*
GOST	*All-Union State Standard*
Goztekhizdat	*State Technical Press*
GTTI	*State Technical and Theoretical Press*
GUPIAE	*State Office for Utilization of Atomic Energy*
IF KhI	*Institute of Physical Chemistry Research*
IFP	*Institute of Physical Problems*
IL	*Foreign Literature Press*
IPF	*Institute of Applied Physics*
IPM	*Institute of Applied Mathematics*
IREA	*Institute of Chemical Reagents*
ISN (Izd. Sov. Nauk)	*Soviet Science Press*
IIaP	*Institute of Nuclear Studies*
Izd	*Press (publishing house)*
KISO	*Solar Research Commission*
LETI	*Leningrad Electrotechnical Institute*
LFTI	*Leningrad Institute of Physics and Technology*
LIM	*Leningrad Institute of Metals*
LITMiO	*Leningrad Institute of Precision Instruments and Optics*
Mashgiz	*State Scientific-Technical Press for Machine Construction Literature*
MATI	*Moscow Aviation Technology Institute*
MGU	*Moscow State University*
Metallurgizdat	*Metallurgy Press*
MOPI	*Moscow Regional Pedagogical Institute*
NIAFIZ	*Scientific Research Association for Physics*
NIFI	*Scientific Research Institute of Physics*
NIIMM	*Scientific Research Institute of Mathematics and Mechanics*
NII ZVUKSZAPIOI	*Scientific Research Institute of Sound Recording*
NIKFI	*Scientific Institute of Motion Picture Photography*
OIIaI	*Joint Institute of Nuclear Studies*
ONTI	*United Scientific and Technical Press*
OTI	*Division of Technical Information*
OTN	*Division of Technical Science*
RIAN	*Radium Institute, Academy of Sciences of the USSR*
SPB	*All-Union Special Planning Office*
Stroiizdat	*Construction Press*
URALFTI	*Ural Institute of Physics and Technology*

NOTE: Abbreviations not on this list and not explained in the translation have been transliterated, no further information about their significance being available to us.—*Publisher.*

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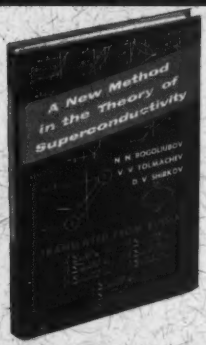

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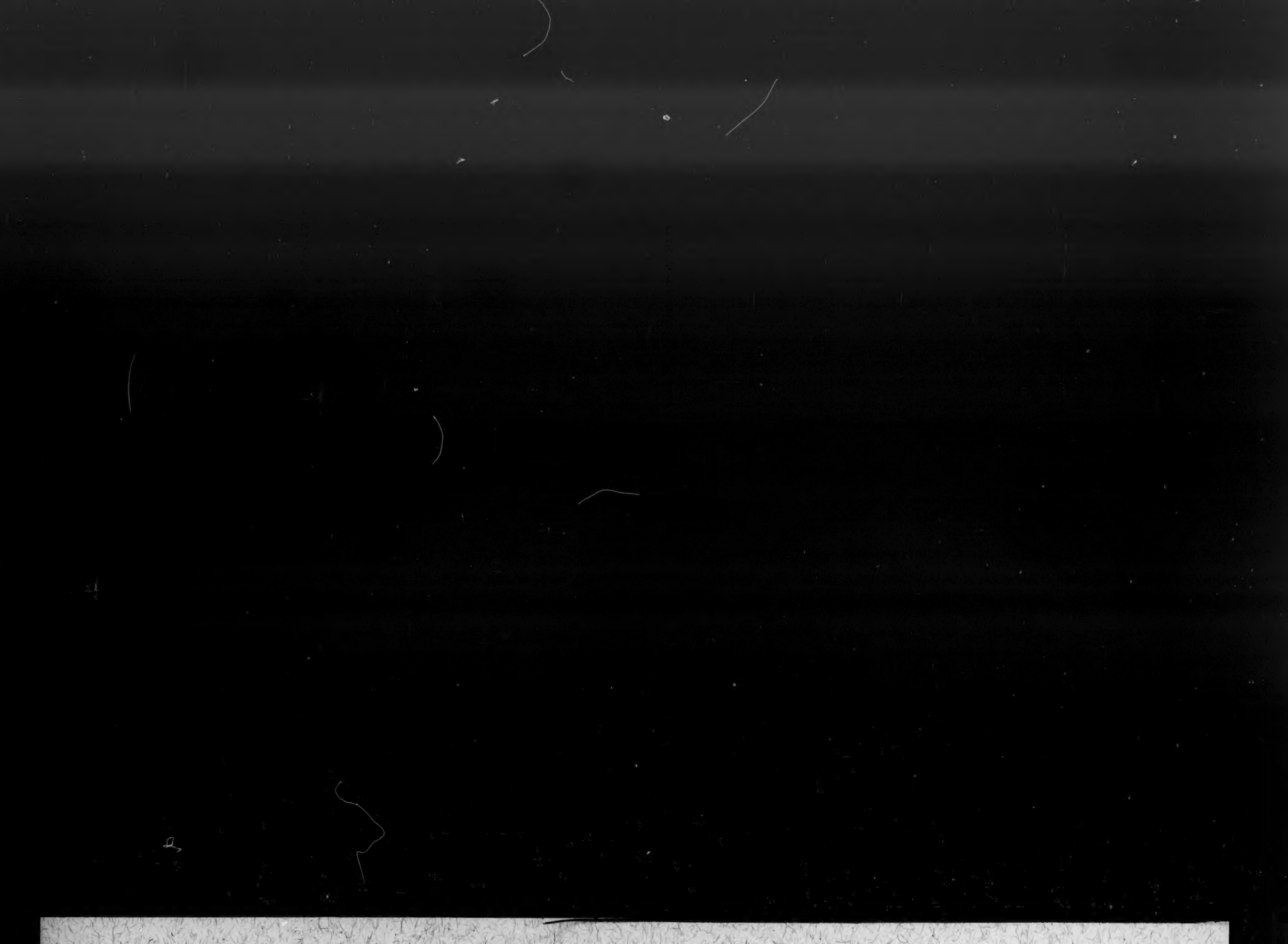
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